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FPL

NOVEMBER 2 1988

L-88-478

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Gentlemen:

Re: Turkey Point Units 3 and 4
Docket Nos. 50-250 and 50-251
Revised Technical Specifications
Proof and Review Comments

The purpose of this letter is to transmit Florida Power & Light Company (FPL) comments on the Proof and Review version of the revised Technical Specifications for Turkey Point Units 3 and 4. This Proof and Review version was issued to FPL with an NRC letter dated June 9, 1988.

The FPL comments are provided in a marked up form in the attachment to this letter. Justification sheets for non-editorial comments are provided, which are indexed to the numbers adjacent to the individual comments.

Further comments, as a result of issues that were identified late in the review process, are expected to be provided during the comment resolution period. These comments are expected to involve control room emergency ventilation and intake cooling water strainers.

The numerical values for items such as pump flows, tank levels, and setpoints are being reviewed through a parallel effort as we have previously discussed. Should any of these values change as a result of our review, we will advise the Technical Specification Branch and the Project Manager.

As previously discussed with the staff, we have not provided comments on Section 3/4.8 - Electrical Power Systems and this section has been deleted from our submittal. A separate license amendment on this subject is being prepared which we will be providing later this month. Upon approval of this amendment, it will replace our previous submittal of Section 3/4.8.

A001
1/1

At our last meeting with the staff a milestone schedule for the remainder of this project was discussed. The period between November 4, 1988 and January 6, 1989 was provided for resolution of comments. Due to the extent and complexity of our comments, we would like to work with the staff to develop a working schedule to detail the process that will be necessary to reach this next milestone. We are prepared to provide the support necessary to meet this milestone. The details of the schedule to meet the January 6, 1989 date can be discussed with the FPL manager for this project, Mr. J. Arias, at (305) 246-6007.

Very truly yours,

W. F. Conway

W. F. Conway
Senior Vice President - Nuclear

WFC/PLP/gp

Attachment

cc: Malcolm L. Ernst, Acting Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

8811100265

PR

SECTION 1.0

DEFINITIONS

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1.7A CONTROLLED LEAKAGE . . . . . 1-2 PR E

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DEFINITIONS SHOULD BE RENUMBERED TO REFLECT ENCLOSED CHANGES.

E

1.0 DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

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

AXIAL FLUX DIFFERENCE ($F_{\Delta I}$)

E

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

JUN 0 9 1983

CONTROLLED LEAKAGE

1.7A CONTROLLED LEAKAGE shall be that seal water flow from the reactor coolant pump seals.

PR #3

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

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

b. The equipment hatch is closed and sealed,

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

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

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

CORE ALTERATIONS

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

DOSE EQUIVALENT I-131

1.9 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microCurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites" or Table E-7 of NRC Regulatory Guide 1.109, Revision 1, October 1977.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.10 $\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (MeV/d) for the radionuclides in the sample isotopes, other than iodines, with half lines greater than 15 minutes, making up at least 95 percent of the total non-iodine activity in the coolant.

PR

DEFINITIONS

FREQUENCY NOTATION

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

GAS DECAY TANK SYSTEM

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

IDENTIFIED LEAKAGE

1.13 IDENTIFIED LEAKAGE shall be:

(except CONTROLLED LEAKAGE)

a. Leakage into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or #3

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

c. Reactor Coolant System leakage through a steam generator to the Secondary Coolant System.

~~MASTER RELAY TEST~~

~~1.14 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.~~ E

MEMBER(S) OF THE PUBLIC

1.15 MEMBER(S) OF THE PUBLIC shall include all persons who are not occupationally associated with the plant. This category does not include employees of the licensee, its contractors, vendors or members of the Armed Forces using property located within the SITE BOUNDARY. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OFFSITE DOSE CALCULATION MANUAL

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

PR

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM

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

PURGE - PURGING

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

PR

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

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

~~REFERENCE POSITION~~

~~1.25 Analog Rod Position Indication System REFERENCE POSITION is defined as:~~

~~a. For all Shutdown Banks and Control Banks A and B; the group demand counter indicated position between 0 and 30 steps withdrawn inclusive and between 200 and 228 steps withdrawn inclusive.~~

~~b. For Control Banks C and D; the group demand counter indicated position between 0 and 30 steps withdrawn inclusive and between 150 and 228 steps withdrawn inclusive. For the withdrawal range of 31 to 149 steps inclusive the REFERENCE POSITION shall be the individual rod calibration curve noting indicated analog rod position versus indicated group demand counter position.~~

#1

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.27 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

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

SLAVE RELAY TEST

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

P R

DEFINITIONS

SOLIDIFICATION

1.30 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping requirements and ~~burial ground requirements.~~ the applicable licensing requirements of the consignee. #4

SOURCE CHECK

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

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

1.35 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE. #3

UNRESTRICTED AREA

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

VENTILATION EXHAUST TREATMENT SYSTEM

1.37 A VENTILATION EXHAUST TREATMENT SYSTEM shall be any system designed and installed to reduce gaseous radioiodine or radioactive material in particulate form in effluents by passing ventilation or vent exhaust gases through charcoal

PR

DEFINITIONS

VENTILATION EXHAUST TREATMENT SYSTEM (Continued)

adsorbers and/or HEPA filters for the purpose of removing iodines or particulates from the gaseous exhaust stream prior to the release to the environment. Such a system is not considered to have any effect on noble gas effluents. Engineered Safety Features Atmospheric Cleanup Systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components. E

VENTING

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

PR

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
~~4/M~~	~~At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year.~~ E (NOT USED)
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 refueling, not to exceed 24 months #5
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

PR

TABLE 1.2

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	% RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ ~~0.90~~ 0.95	0	$\leq$ 140°F) #2

*Excluding decay heat.

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

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PR

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2:

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

MODES 3, 4 and 5:

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

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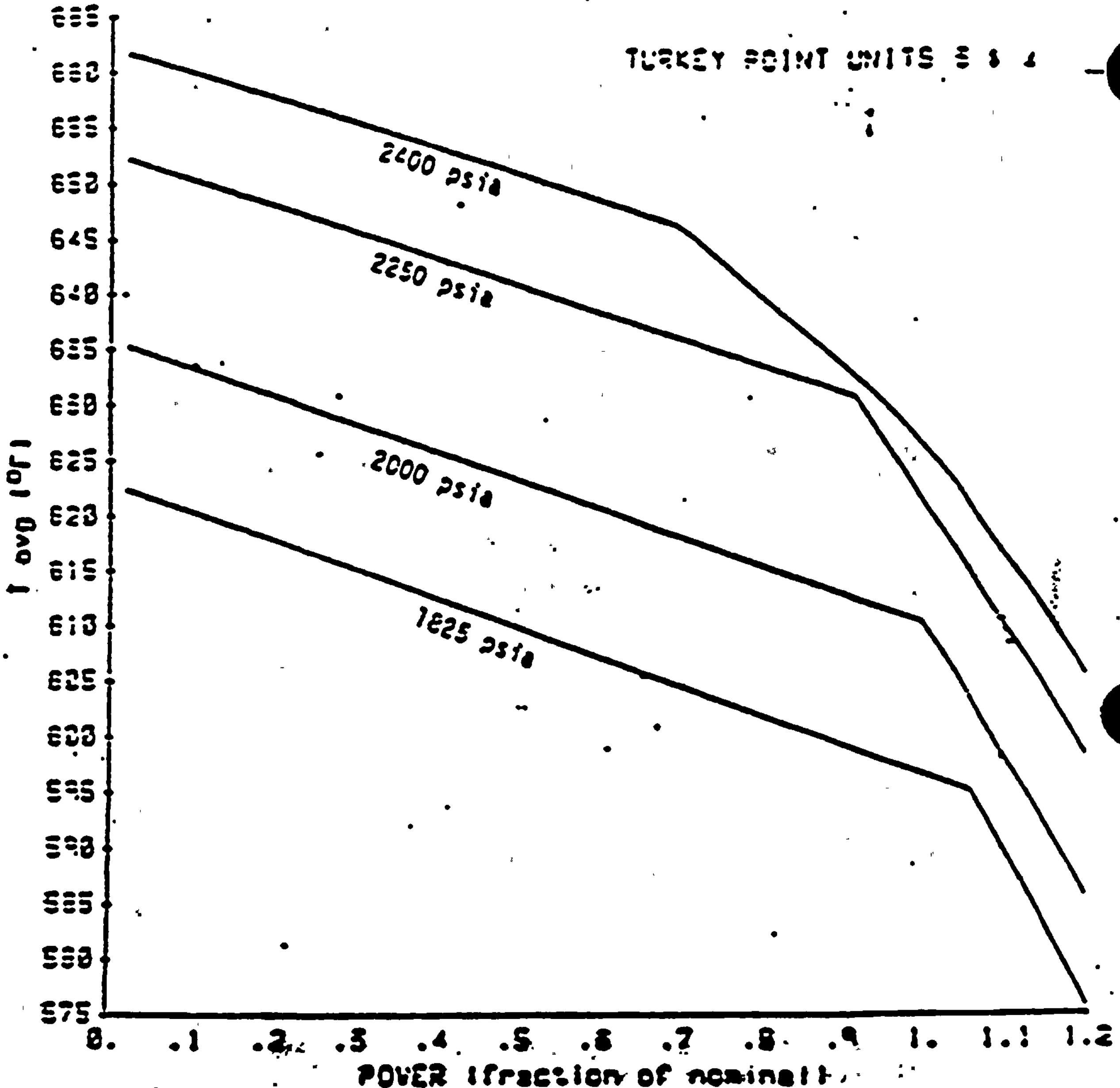

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

PR

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

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

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TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Manual Reactor Trip	N.A.	~~N.A.~~
2. Power Range, Neutron Flux a. High Setpoint	$\leq$109% of RTP**	~~$\leq$110% of RTP**~~
b. Low Setpoint	$\leq$25% of RTP**	~~$\leq$26% of RTP**~~
3. Intermediate Range, Neutron Flux	$\leq$25% of RTP**	~~$\leq$30% of RTP**~~
4. Source Range, Neutron Flux	$\leq 10^5$ cps	~~$\leq 1.3 \times 10^5$ cps~~
5. Overtemperature ΔT	See Note 1	~~See Note 2~~
6. Overpower ΔT	See Note ~~3~~ 2	~~See Note 4~~
7. Pressurizer Pressure-Low	$\geq$1835 psig	~~$\geq$1825 psig~~
8. Pressurizer Pressure-High	$\leq$2385 psig	~~$\leq$2395 psig~~
9. Pressurizer Water Level-High	$\leq$92% of instrument span	~~$\leq$93% of instrument span~~
10. Reactor Coolant Flow-Low	$\geq$90% of loop design flow*	~~$\geq$89% of loop design flow*~~
11. Steam Generator Water Level Low-Low	$\geq$15% of narrow range instrument span	~~$\geq$14% of narrow range instrument span~~

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#1

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

PR

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
12. Steam/Feedwater Flow Mismatch Coincident With	Feed Flow $<0.64 \times 10^6$ lb/hr below steam flow	Feed Flow $<0.72 \times 10^6$ lb/hr below steam flow
Steam Generator Water Level-Low-Low	≥ ~~<~~15% of narrow range instrument span	≤14% of narrow range instrument span
13. Undervoltage - 4.16 kV Busses A and B	≥2496 volts-each bus	≥2456 volts-each bus
14. Underfrequency - Trip of Reactor Coolant Pump Breaker(s) Open	≥56.1 Hz	≥56.0 Hz
15. Turbine Trip		
a. Auto Stop Oil Pressure	≥45 psig	≥40 psig
b. Turbine Stop Valve Closure	Fully Closed #	Fully Closed #
16. Safety Injection Input from ESF	N. A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 1 \times 10^{-10}$ amp	$\geq 6 \times 10^{-11}$ amp

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#1

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

0 9 1986

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TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
b. Low Power Reactor Trips Block, P-7		
1) P-10 ~~input~~ reset	≤10% of RTP**	~~≤11% of RTP**~~
2) Turbine First Stage Pressure	≤10% Turbine Power	~~≤11% Turbine Power~~
c. Power Range Neutron Flux, P-8	≤45% of RTP**	~~≤46% of RTP**~~
d. Power Range Neutron Flux, P-10	≥10% of RTP**	~~≥8% of RTP**~~
18. Reactor Coolant Pump Breaker Position Trip	N.A.	~~N.A.~~
19. Reactor Trip Breakers	N.A.	~~N.A.~~
20. Automatic Trip and Interlock Logic	N.A.	~~N.A.~~

) #4

#1

**RTP = RATED THERMAL POWER

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PR

TABLE 2.2-1 (Continued)

TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

Where:

ΔT = Measured ΔT by RTD Instrumentation;

$\frac{1}{1 + \tau_1 S}$ = Lag compensator on measured ΔT;

τ_1 = Time constants utilized in the lag compensator for ΔT, τ_1 = 2 s;

ΔT_0 = Indicated ΔT at RATED THERMAL POWER;

K_1 = ~~.095;~~ 1.095; | E

K_2 = 0.0107/°F;

$\frac{1 + \tau_2 S}{1 + \tau_3 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_2, τ_3 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_2 = 25s, τ_3 = 3 s;) E

T = Average temperature, °F;

$\frac{1}{1 + \tau_4 S}$ = Lag compensator on measured T_{avg};

τ_4 = Time constant utilized in the measured T_{avg} lag compensator, τ_4 = 2 s;

T' ≤ 574.2°F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = 0.000453/psig;

P = Pressurizer pressure, psig;

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

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

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

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

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

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

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

~~NOTE 2: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2%.~~

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

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NOTE ~~3~~ 2: OVERPOWER ΔT

$$\Delta T \left(\frac{1}{1 + \tau_1 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_5 S}{1 + \tau_5 S}\right)\left(\frac{1}{1 + \tau_4 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_4 S}\right) - T''\right] - f(\Delta I)\right\}$$

Where:

ΔT	=	As defined in Note 1,
$\frac{1}{1 + \tau_1 S}$	=	As defined in Note 1,
τ_1	=	As defined in Note 1,
ΔT_0	=	As defined in Note 1,
K_4	=	1.09,
K_5	=	0.02/°F for increasing average temperature and 0 for decreasing average temperature,
$\frac{\tau_5 S}{1 + \tau_5 S}$	=	The function generated by the rate-lag compensator for T_{avg} dynamic compensation,
τ_5	=	Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_5 = 10$ s,
$\frac{1}{1 + \tau_4 S}$	=	As defined in Note 1,
τ_4	=	As defined in Note 1,

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TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE ~~3~~ 2: (Continued)

K_6 = 0.00068/°F for T > T" and K_6 = 0 for T $\leq$ T",

T = As defined in Note 1,

T" = Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, $\leq$ 574.2°F),

S = As defined in Note 1, and

f (ΔI) = As defined in Note 1.

~~NOTE 4: The channel's maximum Trip Setpoint shall not exceed its computed Trip Setpoint by more than 2.0%~~

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

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2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

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

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB. This relationship has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

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

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

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.62 and and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

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

Where P is the fraction of RATED THERMAL POWER.

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

PR

SAFETY LIMITS

BASES

2.1.1 REACTOR CORE (Continued)

Fuel rod bowing reduces the values of DNB ratio (DNBR). The amount of the DNBR reduction is 4.7% for LOPAR fuel with the L-grid DNB correlation and 5.5% for the OFA fuel with the WRB-1 DNB correlation. The penalties are calculated pursuant to "Fuel Rod Bow Evaluation," WCAP-8691-P-A Revision 1 (Proprietary) and WCAP-8692 Revision 1 (Non-Proprietary). The restrictions of the Core Thermal Hydraulic Safety Limits assure that an amount of DNBR margin greater than or equal to the above penalties is retained to offset the rod bow DNBR penalty.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

and / are designed to

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

B31.1 which with ASME

The entire RCS is hydrotested at 125% (3107 psig) of design pressure, to demonstrate integrity prior to initial operation.

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To accomodate instrument drift that may occur between operational tests, the actual trip setpoint is normally set at a more conservative limit than the setpoint allowed by Table 2.2-1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

~~To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the Reactor Trip Setpoints have been specified in Table 2.2-1. Operation with 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 transient safety analyses.~~

The methodology to derive the Trip Setpoints includes an allowance for instrument uncertainties ~~is based upon combining all of the uncertainties~~ in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensors and other instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. ~~Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.~~ #1

The various Reactor trip circuits automatically open the Reactor trip breakers whenever a condition monitored by the Reactor Trip System reaches a preset or calculated level. In addition to redundant channels and trains, the design approach provides a Reactor Trip System which monitors numerous system variables, therefore providing Trip System functional diversity. The functional capability at the specified trip setting is required for those anticipatory or diverse Reactor trips for which no direct credit was assumed in the safety analysis to enhance the overall reliability of the Reactor Trip System. The Reactor Trip System initiates a Turbine trip signal whenever Reactor trip is initiated. This prevents the reactivity insertion that would otherwise result from excessive Reactor Coolant System cooldown and thus avoids unnecessary actuation of the Engineered Safety Features Actuation System.

Manual Reactor Trip

The Reactor Trip System includes manual Reactor trip capability.

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LIMITING SAFETY SYSTEM SETTINGS

BASES

Power Range, Neutron Flux

In each of the Power Range Neutron Flux channels there are two independent bistables, each with its own trip setting used for a High and Low Range trip setting. The Low Setpoint trip provides protection during subcritical and low power operations to mitigate the consequences of a power excursion beginning from low power, and the High Setpoint trip provides protection during power operations for all power levels to mitigate the consequences of a reactivity excursion which may be too rapid for the temperature and pressure protective trips.

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

Intermediate and Source Range, Neutron Flux

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

Overtemperature ΔT

The Overtemperature ΔT trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the Pressurizer High and Low Pressure trips. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, (2) pressurizer pressure, and (3) axial power distribution. With normal axial power distribution, this Reactor trip limit is always below the core Safety Limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the Reactor trip setpoint is automatically reduced according to the notations in Table 2.2-1.) E

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LIMITING SAFETY SYSTEM SETTINGS

BASES

Overpower ΔT

The Overpower ΔT trip prevents power density anywhere in the core from exceeding 118% of the design power density. This provides assurance of fuel integrity (e.g., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions, limits the required range for Overtemperature ΔT trip, and provides a backup to the High Neutron Flux trip. The setpoint is automatically varied with: (1) coolant temperature to correct for temperature induced changes in density and heat capacity of water, (2) rate of change of temperature for dynamic compensation for piping delays from the core to the loop temperature detectors, and (3) axial power distribution, to ensure that the allowable heat generation rate (kW/ft) is not exceeded.

Pressurizer Pressure

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

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

The High Setpoint trip functions in conjunction with the pressurizer ~~relief and~~ safety valves to protect the Reactor Coolant System against system overpressure. #2

Pressurizer Water Level

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

Reactor Coolant Flow

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

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

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LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Coolant Flow (Continued)

of full power equivalent), an automatic Reactor trip will occur if the flow in more than one loop drops below 90% of nominal full loop flow. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic Reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. Conversely, on decreasing power between P-8 and the P-7 an automatic Reactor trip will occur on low reactor coolant flow in more than one loop and below P-7 the trip function is automatically blocked.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip protects the reactor from loss of heat sink in the event of a sustained steam/feedwater flow mismatch resulting from loss of normal feedwater. The specified setpoint provides allowances for starting delays of the Auxiliary Feedwater System.

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

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

Undervoltage - 4.16 kV Bus A and B Trips

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

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

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

Turbine Trip

~~the~~ reactor trip on

A Turbine trip initiates a Reactor trip. On decreasing power ~~the~~ Turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7. #3

Safety Injection Input from ESF

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

Reactor Coolant Pump Breaker Position Trip

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

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

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LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Trip System Interlocks

The Reactor Trip System interlocks perform the following functions:

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

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

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

P-10 On increasing power, P-10 allows the manual block of the Intermediate Range trip and the Low Setpoint Power Range trip; and automatically blocks the Source Range trip and deenergizes the Source Range high voltage power. On decreasing power, the Intermediate Range trip and the Low Setpoint Power Range trip are automatically reactivated. P-10 also provides input to P-7.

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITIONS FOR OPERATION

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

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

3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within 1 hour action shall be initiated to place it, as applicable, in:

a. At least HOT STANDBY within the next 6 hours,* #1

b. At least HOT SHUTDOWN within the following 6 hours, and

c. At least COLD SHUTDOWN within the subsequent 24 hours.

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

This specification is not applicable in MODES 5 or 6.

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

*Except when the ACTION applies to both units simultaneously, then both units shall be in at least HOT STANDBY within 12 hours. #1

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APPLICABILITY

LIMITING CONDITIONS FOR OPERATION (Continued)

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

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

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

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

3.0.6. When a system, subsystem, train, component or device is determined to be inoperable soley because its emergency power source is inoperable, or solely because its normal power source is inoperable, it may be considered OPERABLE for the purpose of satisfying the requirements of its applicable Limiting Condition for Operation, provided: (1) its corresponding normal or emergency power source is OPERABLE; and (2) all of its redundant system(s), subsystem(s), train(s), component(s) and device(s) are OPERABLE, or likewise satisfy the requirements of this specification. This specification is not applicable in MODES 5 or 6.

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

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

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APPLICABILITY

SURVEILLANCE REQUIREMENTS (CONTINUED)

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

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

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

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

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

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

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

RCS Average Temperature

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

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

*See Special Test Exceptions Specification 3.10.1.

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SURVEILLANCE REQUIREMENTS (Continued)

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

4.1.1.1.2 When in MODE 1 or 2, the overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1% Δk/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e, above. The predicted reactivity values ~~shall~~ may be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPD after each fuel loading. #3

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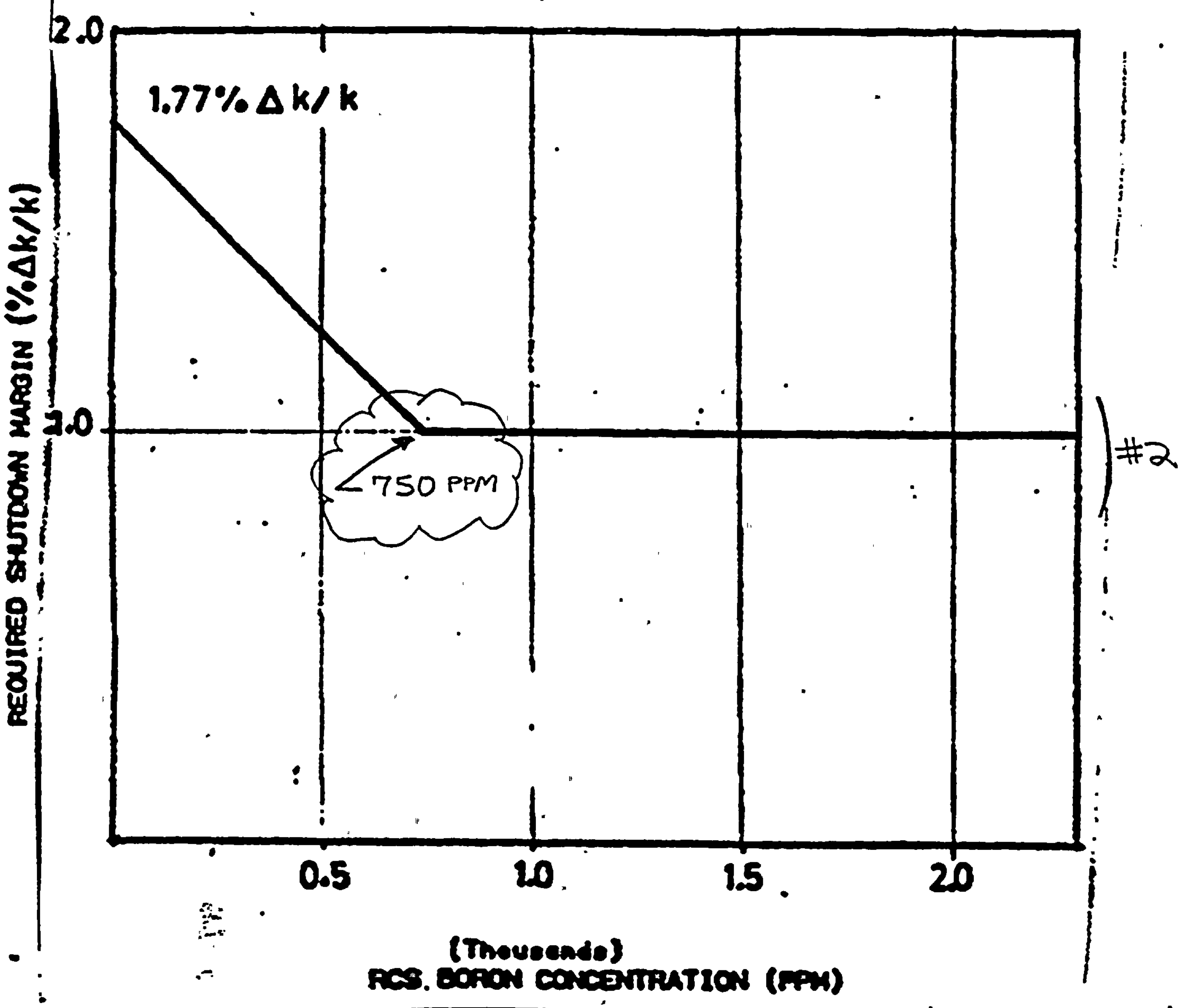

Figure 3.1-1
Required Shutdown Margin vs Reactor Coolant
Boron Concentration

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SHUTDOWN MARGIN - RCS Average Temperature ~~T_{avg}~~ LESS THAN OR EQUAL TO 200°F)E

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 1% $\Delta k/k$, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or ~~equivalent~~ until the required SHUTDOWN MARGIN is restored.

up to the flow capacity of one OPERABLE charging pump with suction from an OPERABLE RWST #1

SURVEILLANCE REQUIREMENTS

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

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

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

1) Reactor Coolant System boron concentration,

2) Control rod position,

3) Reactor Coolant System average temperature,

4) Fuel burnup based on gross thermal energy generation,

5) Xenon concentration, and

6) Samarium concentration.

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REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

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

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

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

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

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

ACTION:

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

1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive or equal to limits described in 3.1.1.3a, b and c above within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6;

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

3. A Special Report is prepared and submitted to the Commission, pursuant to Specification 6.9.2, within ~~10~~ 30 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. #1

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

**See Special Test Exceptions Specification 3.10.3.

PR

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

PR

REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2* **.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 547°F with the T_{avg}-T_{ref} Deviation Alarm not reset.

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

**See Special Test Exceptions Specification 3.10.3.

PR

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 storage tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.5a. 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 ~~12 hours~~ 7 days by verifying that the temperature of the heat traced portion of the flow path is greater than or equal to 145[illegible] when a flow path from the boric acid tanks is used, and #1

b. At least once per 31 days by verifying that each accessible* valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. #2

*· Locked high radiation areas (e.g., containment at power) are considered inaccessible. #2

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

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

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

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

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

ACTION:

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

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

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

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

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

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

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REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS

4.1.2.2 The above required ~~flow paths~~ shall be demonstrated OPERABLE:

a. At least once per ~~12 hours~~ 7 days by verifying that the temperature of the heat traced portion ~~of the~~ flow path from the boric acid tanks is greater than or equal to 145°F when it is a required ~~water~~ source; #1

b. At least once per 31 days by verifying that each (accessible**) valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position; #2

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. and c. delivers at least 10 gpm to the RCS.

** Locked high radiation areas (e.g., containment at power) are considered inaccessible. #2

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3, and 4

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A Boric Acid Storage System with:

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

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

3) A minimum solution temperature of 145°F.

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

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

2) A minimum boron concentration ~~between~~ of 1950 ppm and

3) A minimum solution temperature of 39°F.

E

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the indicated ~~contained~~ borated water volume, and

~~b) At least once per 12 hours by~~ 3) Verifying the boric acid storage tank solution temperature when it is the source of borated water.

E

#1

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

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

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

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REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.5 The following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

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

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

3) A minimum solution temperature of 145°F.

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

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

2) A minimum boron concentration of 1950 ppm,

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

4) A maximum solution temperature of 100°F.

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

ACTION:

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

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

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4.1.2.5 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

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

b. At least once per ~~12 hours~~ 7 days by verifying the Boric Acid Storage System solution temperature when it is the source of borated water. #1

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

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

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

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HEAT TRACING

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

MODES 1, 2, 3 and 4

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

MODES 5 and 6

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 7 days ~~12 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. #1

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REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

Analog Rod Position Indication | E

3.1.3.1 All full length (shutdown and control) rods shall be OPERABLE and positioned within ± 12 steps (~~indicated position~~) of the ~~REFERENCE POSITION corresponding to the~~ group step counter demand position within one hour after rod motion. | #1

APPLICABILITY: MODES 1* and 2*

ACTION:

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

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

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

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

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

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

*See Special Test Exceptions 3.10.2 and 3.10.3.

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SWITCH LOCATIONS

d ~~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, | E

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

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

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

SURVIELLANCE REQUIREMENTS

| E

4.1.3.1.1 The position of each full length rod shall be determined to be within ± 12 steps (~~indicated position~~) counter Analog Rod Position Indication of the ~~REFERENCE POSITION corresponding to the~~ group step demand position at least once per 12 hours (allowing for one hour thermal soak after rod motion) except during time invervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours. | #1

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

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TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

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

Single Rod Cluster Control Assembly Withdrawal at Full Power

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

Major Secondary Coolant System Pipe Rupture

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

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REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS - OPERATING

LIMITING CONDITION FOR OPERATION

The analog rod position indication system and the demand position indication system

3.1.3.2 ~~The shutdown and control rod position indication systems~~ shall be OPERABLE and capable of determining the actual and demanded control rod positions respectively as follows:

E

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

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

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

Control Banks C and D: ± 12 steps of the group demand counters for withdrawal range of 0-~~30 steps and 150~~-228 steps. ~~± 12 steps of the REFERENCE POSITION for withdrawal range of 31-149 steps.~~ #2

b. Group demand counters; ± 2 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and within one hour after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

2. Reduce THERMAL POWER to less than ~~50%~~ 75% of RATED THERMAL POWER within 8 hours. #1

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

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

2. Reduce THERMAL POWER to less than ~~50%~~ 75% of RATED THERMAL POWER within 8 hours. #1

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SURVEILLANCE REQUIREMENTS

4.1.3.2.1 Each analog rod position indicator shall be determined to be OPERABLE by verifying that the demand position indication system and the rod position indication system ~~(by use of the REFERENCE POSITION)~~ agree within 12 steps (allowing for one hour thermal soak after rod motion) at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indication system ~~(by use of the REFERENCE POSITION)~~ at least once per 4 hours. #2 #2

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

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TABLE 4.1-1

ROD POSITION INDICATOR SURVEILLANCE REQUIREMENTS

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

E

~~*Not required during Cold Shutdown~~

E

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POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

---

*With the Reactor Trip System breakers in the closed position.
#See Special Test Exceptions Specification 3.10.~~5~~.4 E

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ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.4 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. For specifically affected individual rods following any maintenance on or modification to the Control Rod Drive System which could affect the drop time of those specific rods, and

c. At least once per ~~18 months.~~ refueling, not to exceed 24 months.

|#1

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REACTIVITY CONTROL SYSTEMS

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be fully withdrawn.

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 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: ~~by use of the group step demand counters, and verified by the analog rod position indicators within one hour after rod motion;~~ #1

a. Within 15 minutes prior to withdrawal of any rods in control banks A, B, C, or D during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

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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 Figures 3.1-2.

APPLICABILITY: MODES 1* and 2*#

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each control bank shall be determined to be within the insertion limits at least once per 12 hours ~~by use of the group step demand counters and verified by the analog rod position indicators within one hour of rod motion~~, except during time intervals when the Rod Insertion Limit Monitor is inoperable, then verify the individual rod positions at least once per 4 hours. #1

*See Special Test Exceptions Specifications 3.10.2 and 3.10.3.

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

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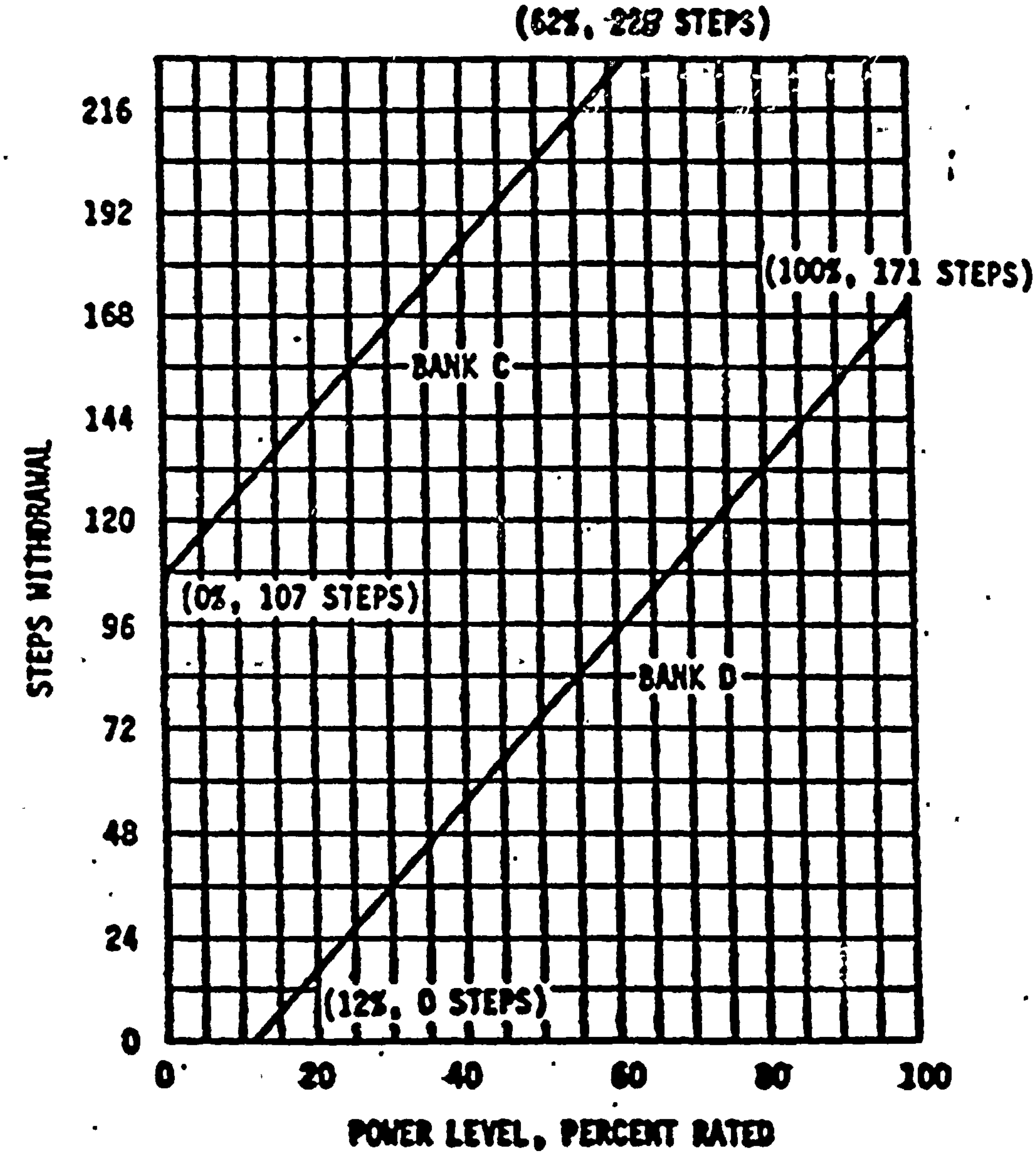

FIGURE 3.1-2

ROD BANK INSERTION LIMITS VERSUS THERMAL POWER
THREE LOOP OPERATION

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

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

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

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

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

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

*See Special Test Exceptions Specification 3.10.2.

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

Add Footnote to 4.2.1.1.a.2)

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

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c. With the indicated AFD outside of the above required target band for more than 1 hour of cumulative penalty deviation time during the previous 24 hours and with THERMAL POWER less than 50% but greater than 15% of RATED THERMAL POWER, the THERMAL POWER shall not be increased equal to or greater than 50% of RATED THERMAL POWER until the indicated AFD is within the above required target band.

SURVEILLANCE REQUIREMENTS

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

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

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

   2) At least once per hour for the first ~~24~~ 6 hours after restoring the alarm used to monitor the AFD to OPERABLE status.*

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

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

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

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

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

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

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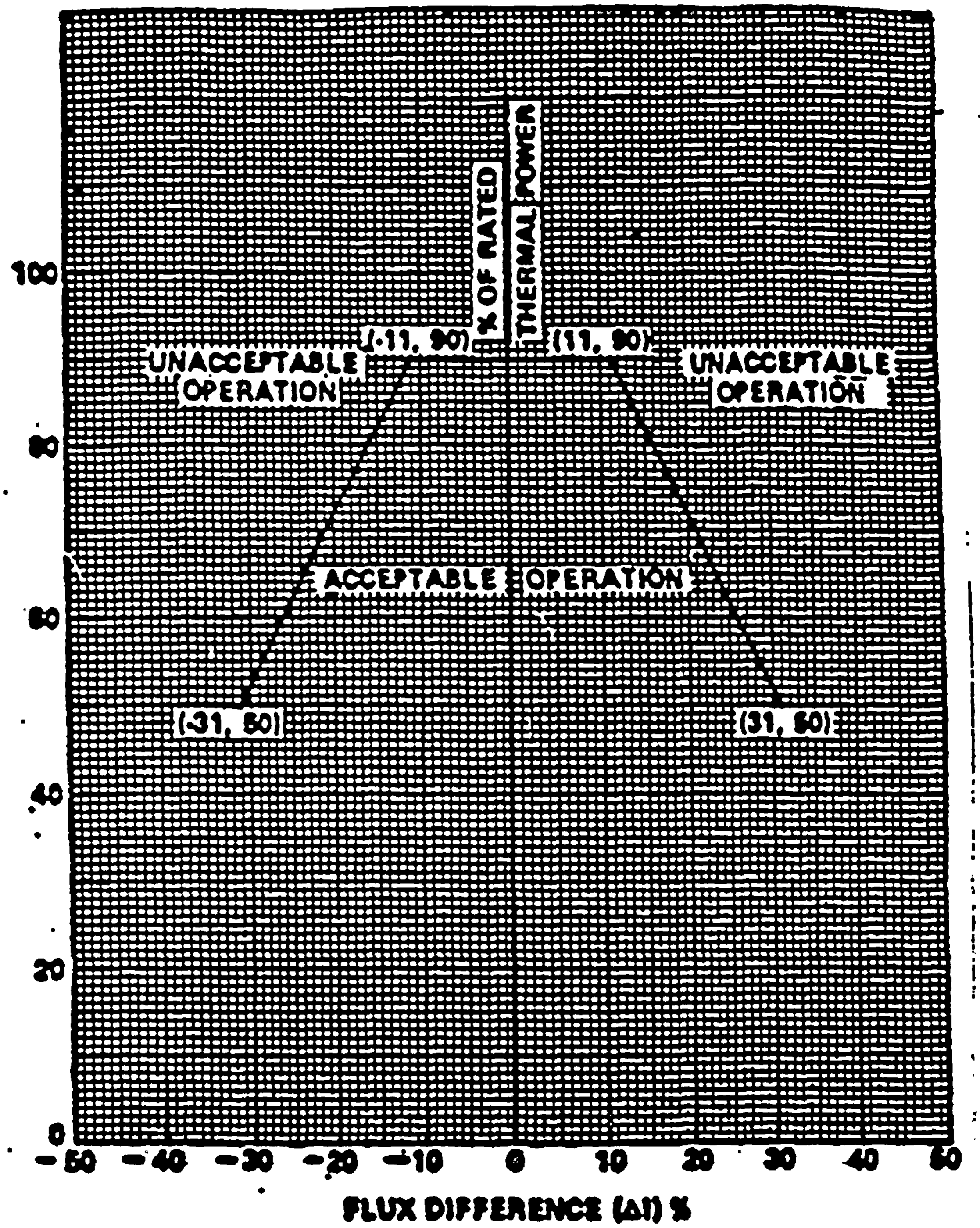

FIGURE 3.2-1

AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF
RATED THERMAL POWER

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POWER DISTRIBUTION LIMITS

REPLACE WITH FPL WORDING

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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:

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

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

where:

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

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

APPLICABILITY: MODE 1

ACTION:

With $F_Q(Z)$ exceeding its limit:

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

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

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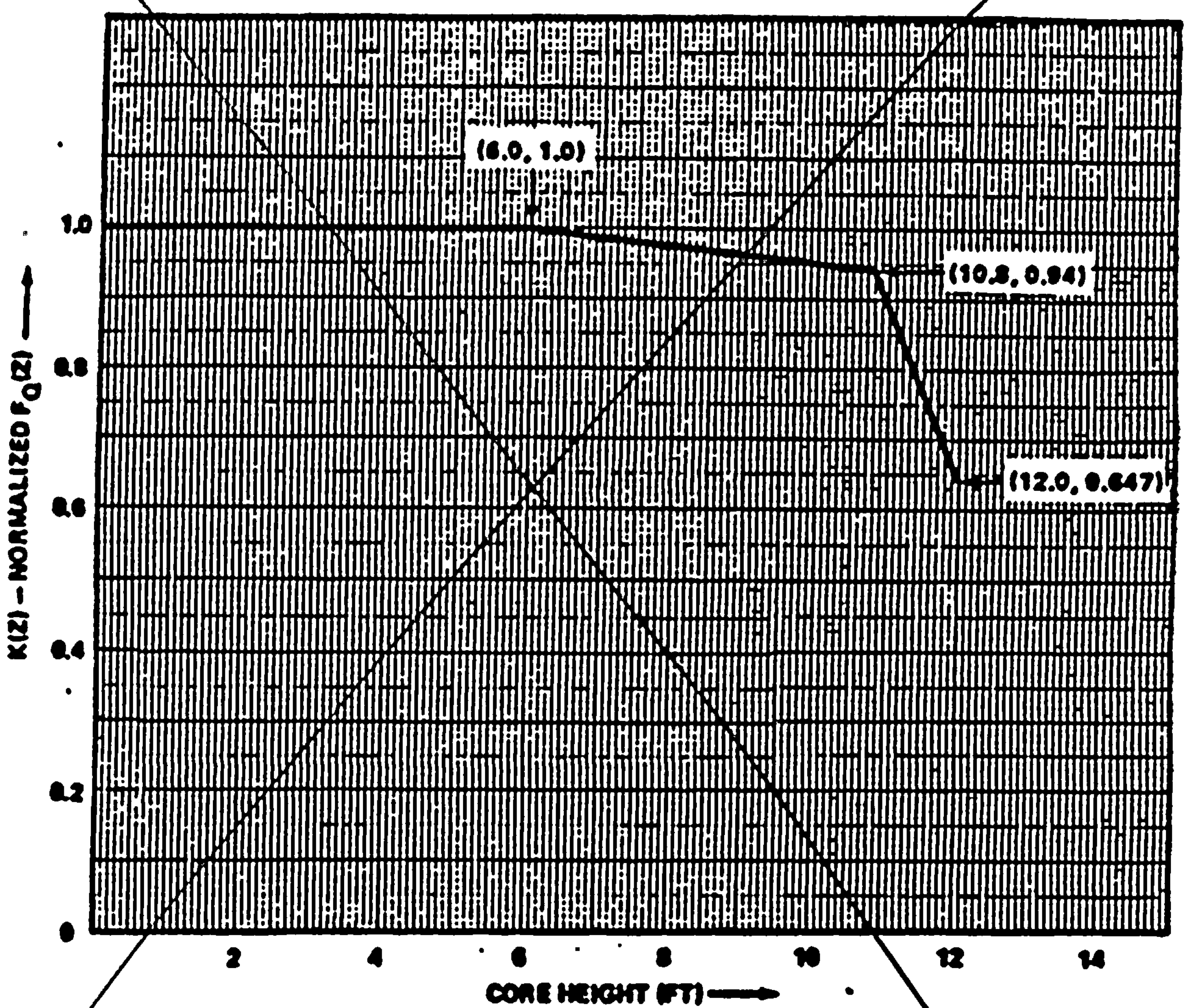

FIGURE 3.2-2
K(Z) NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

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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 Specification 4.2.2.2b., above to:

1) The F_{xy} limits for RATED THERMAL POWER (F_{xy}^{RTP}) for the appropriate measured core plans given in Specification 4.2.2.2.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 either:

a) 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 Effective Full Power Days (EFPD), whichever occurs first.

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REPLACE WITH FPL WORDING

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 6.9.1.6;

f. The F_{xy} limits of Specification 4.2.2.2e., 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, and

4) Core plane regions within ± 2% of core height [± 2.88 inches] about the bank demand position of the Bank "D" control rods.

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

1) The $F_Q(Z)$ 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.

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

LIMITING CONDITION FOR OPERATION

3.2.2 $F^L_Q(Z)$ shall be limited by the following relationships:

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

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

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

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

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

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

APPLICABILITY: MODE 1

ACTION:

With the measured value of $F^M_Q(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F^M_Q(Z)$ exceeds $F^L_Q(Z)$ within 15 minutes and similarly reduce the Power Range Neutron Flux - High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta-T Trip Setpoints (value of K_4) have been reduced at least 1% for each 1% $F^M_Q(Z)$ exceeds the $F^L_Q(Z)$; and

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

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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2) The following action shall be taken:

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

4.2.2.2 MIDS

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

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

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

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

* $[F_j(z)]_S$ is the alarm setpoint for MIDS

** P_L is reactor thermal power expressed as a fraction of 1 that is used to calculate $[F_j(z)]_S$

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SURVEILLANCE REQUIREMENTS (Continued)

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

1. At least once every 24 hours, and

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

   1) Raising the thermal power above P_T, or

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

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

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

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

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

where: $\bar{R}$ = Total peaking factor from a full flux map ratioed to the axial peaking factor in a selected thimble.
j = the thimble location selected for monitoring

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4.2.2.3 Base Load

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

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

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

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

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

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

d) Calculate P_{BL} per 4.2.2.3b.

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

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

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

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

d) Calculate P_{BL} per 4.2.2.3b.

b. Base Load operation is permitted provided:

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

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

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

P_{BL} and P_T are defined as:

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

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

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

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

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

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

4.2.2.4 RADIAL BURNDOWN

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

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

$[F_Q(Z)]_{RB}$ Meas. = $[F_{xy}(Z)]_{Map}$ Meas. x $F_z(Z)$ x 1.09 and

$[F_Q^L(Z)]$ is equal to $[F_Q^L]$ x $K(Z)$.

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

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THIS IS SAME
AS P&R FIGURE

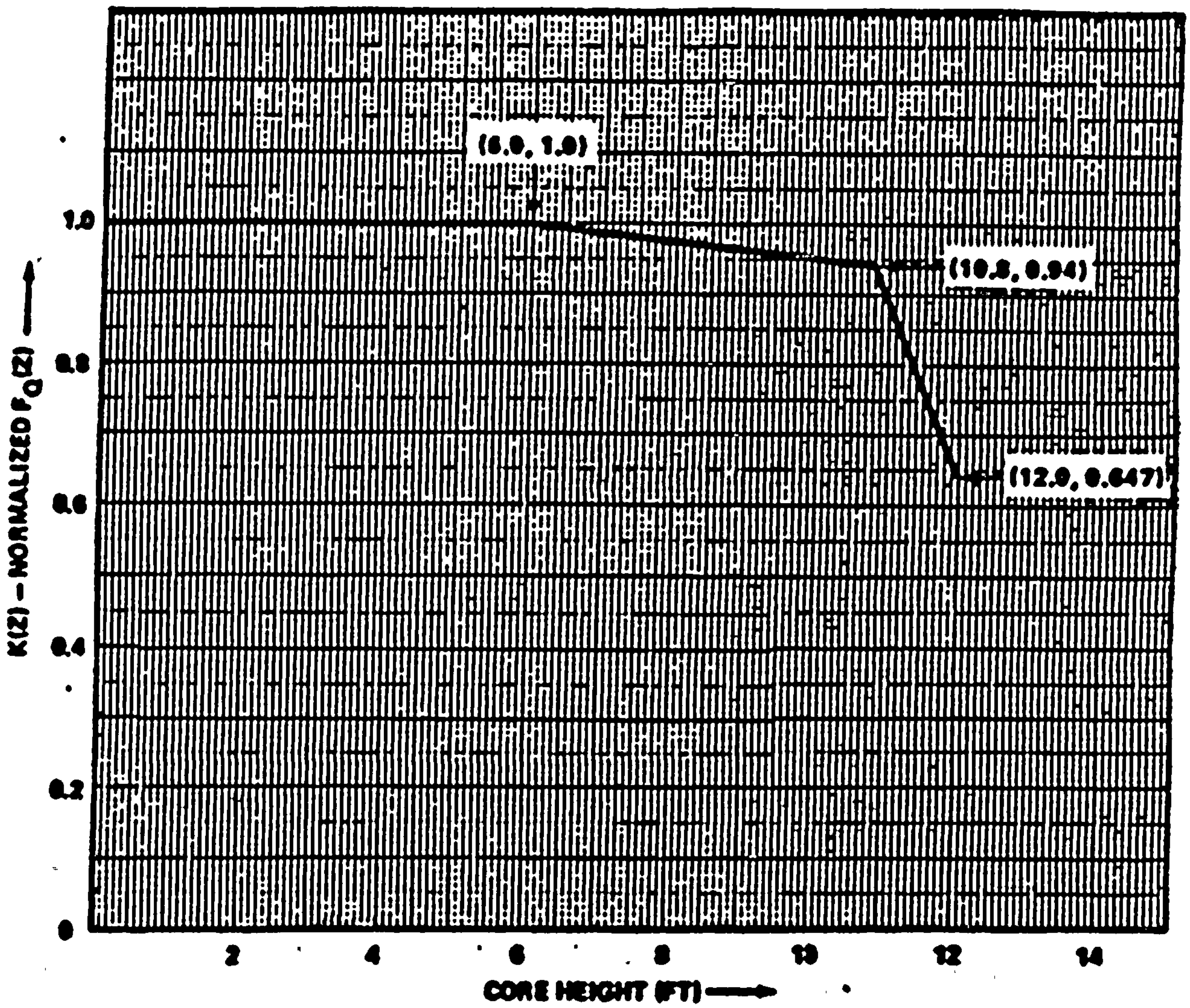

FIGURE 3.2-2
K(Z) NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

4.2.2.5 When $F_Q(Z)$ is measured for reasons other than meeting the requirements of specification 4.2.2.1, 4.2.2.2, 4.2.2.3 or 4.2.2.4 an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

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3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

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

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

Where:

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

APPLICABILITY: MODE 1.

ACTION:

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

a. Within 2 hours either:

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

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

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated, through incore flux mapping, to be within the limit of acceptable operation prior to exceeding the following THERMAL POWER levels:

   1. A nominal 50% of RATED THERMAL POWER,

   2. A nominal 75% of RATED THERMAL POWER, and

   3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

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SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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3/4.2.4 QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 The QUADRANT POWER TILT RATIO shall not exceed 1.02.

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

ACTION:

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

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

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

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

   2. Within 2 hours either:

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

      b) Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

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

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*See Special Test Exceptions Specification 3.10.2.

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

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

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

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

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

   3. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours; and

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

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

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

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

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

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POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Calculating the ratio at least once per 7 days when the Power Range Upper Detector High Flux Deviation and Power Range Lower Detector High Flux Deviation Alarms are OPERABLE, and

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75% of RATED THERMAL POWER with one Power Range channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained either from two sets of four symmetric thimble locations or full-core flux map, or by incore thermocouple map is consistent with the indicated QUADRANT POWER TILT RATIO at least once per ~~12~~ 24 hours. }#2 |#1

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

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POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

		Limit
a.	Reactor Coolant System T_{avg}	$\leq 576.3°F$
b.	Pressurizer Pressure, and	≥ 2222 psig *
c.	Reactor Coolant System Flow	$\geq 277{,}900$ gpm

E

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of 3.2.5.a and 3.2.5.b above ~~Table 3.2-1~~ shall be verified to be within its limits at least once per 12 hours. E

4.2.5.2 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per ~~18 months.~~ refueling, not to exceed 24 months #

4.2.5.3 The RCS flow rate shall be demonstrated by measurement once per ~~18 months.~~ refueling, not to exceed 24 months

* Limit not applicable during either a Thermal Power ramp in excess of 5% of Rated Thermal Power per minute or a Thermal Power step in excess of 10% of Rated Thermal Power E

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TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS
Indicated Reactor Coolant System T_{avg}	$\leq$ 576.3°F
Indicated Pressurizer Pressure	$\geq$ 2287 psia*
Indicated Reactor Coolant Flow	$\geq$ 277,900 gpm~~**~~

E

*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.

~~**Includes a 3.5% flow measurement uncertainty.~~

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the Reactor Trip System instrumentation channels and interlocks of Table 3.3-1 shall be OPERABLE.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each Reactor Trip System instrumentation channel and interlock and the automatic trip logic shall be demonstrated OPERABLE by the performance of the Reactor Trip System Instrumentation Surveillance Requirements specified in Table 4.3-1.

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
1. Manual Reactor Trip	2	1	2	1, 2	1	
	2	1	2	3*, 4*, 5*	9	
2. Power Range, Neutron Flux						
a. High Setpoint	4	2	3	1, 2	2	
b. Low Setpoint	4	2	3	1##, 2	2	
3. Intermediate Range, Neutron Flux	2	1	2	1##, 2	3	
4. Source Range, Neutron Flux						
a. Startup	2	1	2	2#	4	
b. Shutdown	2~~**~~	0	2~~**~~	3**, 4**, 5**	5	)E
c. Shutdown	2	1	2	3*, 4*, 5*	9	
5. Overtemperature ΔT	3	2	2	1, 2	6	
6. Overpower ΔT	3	2	2	1, 2	6	
7. Pressurizer Pressure--Low (Above P-7)	3	2	2	1	6	)#1-1
8. Pressurizer Pressure--High	3	2	2	1, 2	6	
9. Pressurizer Water Level--High (Above P-7)	3	2	2	1	6	)#1
10. Reactor Coolant Flow--Low						
a. Single Loop (Above P-8)	3/loop	2/loop	2/loop	1	6	
b. Two Loops (Above P-7 and below P-8)	3/loop	2/loop	2/loop	1	6	

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
11. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen.	2/stm. gen.	1, 2	6# 2	)E
12. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feedwater flow mismatch in each stm. gen.	1 stm. gen. level coincident with 1 stm./feedwater flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feedwater flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6	
13. Undervoltage--4.16 KV Busses A and B (Above P-7)	2/bus	1/bus on both busses	2/bus	1	13	)#1
14. Underfrequency--Trip of Reactor Coolant Pump Breaker(s) Open (Above P-7)	2/bus	1 to trip RCPs***	2 ~~1~~/bus]#9	1	11	)#1
15. Turbine Trip						
a. Autostop Oil Pressure	3	2	2	1	13	
b. Turbine Stop Valve Closure (Above P-7)	2	2	2	1	13	)#1

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TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION	
16. Safety Injection Input from ESF	2	1	2	1, 2	8	
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux~~,~~ P-6 (decreasing power)	2	~~1~~ 2) #11	2	2#	7	#2
b. Low Power Reactor Trips ~~Block~~, P-7 (increasing power)						
Enable P-10 ~~Input~~ reset	4	2	3	1	7	E
or						
Turbine First Stage Pressure	2	1	~~2~~ 1) #10	1	7	
c. Power Range Neutron Flux~~,~~ P-8 (increasing power)	4	2	3	1	7	#2
d. Power Range Neutron Flux~~,~~ P-10 (decreasing power)	4	~~2~~ 3) #12	3	1, 2	7	#2
18. Reactor Coolant Pump Breaker Position Trip						
a. Above P-8	1/breaker	1	1/breaker	1	11	
b. Above P-7 and below P-8	1/breaker	2	1/breaker	1	11	
19. Reactor Trip Breakers	2	1	2	1, 2	8, 10	
	2	1	2	3*, 4*, 5*	9	
20. Automatic Trip and Interlock Logic	2	1	2	1, 2	12	
	2	1	2	3*, 4*, 5*	9	

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TABLE 3.3-1 (Continued)

TABLE NOTATIONS

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

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

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

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

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

ACTION STATEMENTS

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

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

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

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

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

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

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

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

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

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

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker to OPERABLE status.

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

ACTION 12 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1, provided the other channel is OPERABLE. Upon determination of an inoperable relay, the associated channel may be out of service for up to 8 hours for repair and testing.

ACTION 13 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed until performance of the next required ACTUATION LOGIC TEST provided the inoperable channel is placed in the tipped condition within 1 hour. Pulling fuses and placing jumpers may be used to place the channel in trip condition. #7

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

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

)E

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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
17. Reactor Trip System Interlocks (Continued)						
d. Power Range Neutron Flux, P-10	N.A.	R(4)	M(8)	N.A.	N.A.	1, 2
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	N.A.	R	N.A.	1
19. Reactor Trip Breaker	N.A.	N.A.	N.A.	M(7, ~~11~~ 13), R(11)	N.A.	1, 2, 3*, 4*, 5* #4
20. Automatic Trip and Interlock Logic	N.A.	N.A.	N.A.	N.A.	M(7, ~~15~~ 14)	1, 2, ~~3*, 4*, 5*~~ #6
~~21. Reactor Trip Bypass Breaker~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~M(14), R(16)~~	~~N.A.~~	~~1, 2, 3*, 4*, 5*~~ #5

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS

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

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

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

(1) If not performed in previous 7 days.

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

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

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

(5) This table Notation number is not used.

(6) Incore-Excore Calibration, above 75% of RATED THERMAL POWER (RTP). If the quarterly surveillance requirement coincides with sustained operation between 30% and 75% of RTP, calibration shall be performed at this lower power level. ~~However, any power increase of greater than 20% within the operating band of 30% to 75% of RTP requires a new calibration to ensure that calibration is maintained within 20% of sustained operating level. In addition, a new calibration shall be performed upon an increase in power to greater than 75% of RTP.~~ The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1. 8

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

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

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

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TABLE 4.3-1 (Continued)

TABLE NOTATIONS (Continued)

(10) Setpoint verifica~~tion is no~~t applicable.

(11) At least once per ~~18 months~~ refueling and following maintenance or adjustment of the Reactor trip bre~~akers, the TR~~IP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the Undervoltage and Shunt trips. #3

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

(13) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip circuits for the Manual Reactor Trip Function. The test shall also verify the OPERABILITY of the Bypass Breaker trip circuit(s).

~~(14) Remote manual undervoltage trip when breaker placed in service.~~ #5

14 (~~15~~) Interlock Logic Test shall consist of verifying that the interlock is in its required state by observing the permissive annunciator window. E

~~(16) Automatic undervoltage trip.~~ #5

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INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-2.

ACTION:

~~a. With an ESFAS Instrumentation or Interlock Trip Setpoint trip less conservative than the value shown in the Trip Setpoint column but more conservative than the value shown in the Allowable Value column of Table 3.3-3, adjust the Setpoint consistent with the Trip Setpoint value.~~ #1

a ~~b.~~ With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in ~~the Allowable Value column of~~ Table 3.3-3, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-2 until the channel is restored to OPERABLE status with its Setpoint adjusted consistent with the Trip Setpoint value. #1

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3.2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water). e					E
a. Manual Initiation	2	1	2	1, 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Containment Pressure--High 1 e	3	2	2	1, 2, 3	15* e E
d. Pressurizer Pressure--Low	3	2	2	1, 2, 3#	15
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1, 2, 3*	15
f. High Steam Line Flow-Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Low Steam ~~Line~~ Generator Pressure ~~or~~ or	1/steam ~~line~~ generator	1/steam ~~line~~ generator in any two steam lines	1/steam ~~line~~ generator in any two steam lines	1, 2, 3*	15 E
Low T_{avg}	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	15

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TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
c. Containment Pressure--High-High Coincident with:	3	2	2	1, 2, 3	15* E
Containment Pressure--High	3	2	2	1, 2, 3	15
d. High Steam Line Flow--Coincident with:	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3*	15
Low Steam ~~Line~~ Generator Pressure	1/steam ~~line~~	1/steam ~~line~~ generator in any two steam lines	1/steam ~~line~~ in any two steam lines	1, 2, 3* #11	15 E
or Low T_{avg}	1/loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3*	15
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
6. Auxiliary Feedwater###					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

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

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TABLE 3.3-2 (Continued)

TABLE NOTATIONS

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

##Channels are R11 for gaseous radioactivity and R12 for particulate radioactivity.)E

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

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

~~**See Bases for Partial Auxiliary Feedwater Start.~~)E

***Applicable during MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment or spent fuel pool.)E

ACTION STATEMENTS

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

ACTION 15 - With the number of OPERABLE channels one ~~less than the Total Number of Channels~~, operation may proceed until performance of the next required ANALOG CHANNEL OPERATIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour. Pulling fuses and placing jumpers may be used to place the channel in trip condition.)#4)#3

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

ACTION 17 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*** Applicable during Modes 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment.)#14

TABLE 3.3-2 (Continued)

ACTION STATEMENTS (Continued)

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

ACTION 19 - With less than the Minimum Number of Channels OPERABLE, within 1 hour determine by observation of the associated permissive annunciator window(s) that the interlock is in its required state for the existing plant condition, or apply Specification 3.0.3.

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

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

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

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

PR

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

#1 Q

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
1. Safety Injection (Reactor Trip, Feedwater Isolation, Control Room Isolation, Start Diesel Generators, Containment Cooling Fans, and Essential Service Water)		E
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. Containment Pressure--High	≤~~4~~ 6 psig #7	≤4.5 psig
d. Pressurizer Pressure--Low	≥~~1725~~ 1715 psig #7	≥1713 psig
e. High Differential Pressure Between the Steam Line Header and any Steam Line.	≤150 psi	≤162 psi
f. High Steam Line Flow--	≤A function defined as follows: A Δp corresponding to 0.64 x 10⁶ lbs/hr at 0% load increasing linearly to a Δp corresponding to 3.84 x 10⁶ lbs/hr at full load	≤A function defined as follows: A Δp corresponding to 0.76 x 10⁶ lbs/hr at 0% load increasing linearly to a Δp corresponding to 3.96 x 10⁶ lbs/hr at full load. E
Generator		
Coincident with: Low Steam ~~Line~~ Pressure--	≥600 psig	≥580 psig E
or Low T_{avg}--	≥~~535~~ 531 °F #7	≥531°F
2. Containment Spray		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Containment Pressure--High-High	≤30.0 psig	≤32.0 psig
Coincident with: Containment Pressure--High	≤~~4~~ 6.0 psig #7	≤4.5 psig

PR

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1 above for all Safety Injection Trip Setpoints, ~~and Allowable Values.~~ #1	
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Containment Pressure--High-High Coincident with: Containment Pressure--High	≤30.0 psig ≤~~4~~6.0 psig #7	≤32.0 psig ≤4.5 psig
c. Containment Ventilation Isolation		
1) Containment Isolation Manual Phase A or Phase B	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints, ~~and Allowable Values.~~	
4) High Containment Radioactivity (1)	Particulate (R-11) ≤6.1 x 10^5 CPM Gaseous (R-12) See (2)	~~(Same as Trip Setpoint)~~ E

#1 #1

PR

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS #1

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
4. Steam Line Isolation		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High Coincident with: Containment Pressure--High	≤30.0 psig ≤~~4~~6.0 psig #7	≤32.0 psig ≤4.5 psig
d. High Steam Line Flow E	≤A function defined as follows: A Δp corresponding to 0.64 x 10^6 lbs/hr at 0% load increasing linearly to a Δp corresponding to 3.84 x 10^6 lbs/hr at full load.	≤A function defined as follows: A Δp corresponding to 0.76 x 10^6 lbs/hr at 0% load increasing linearly to a Δp corresponding to 3.96 x 10^6 lbs/hr at full load.
Coincident with: Generator Low Steam ~~Line~~ Pressure-- or Low T_{avg} E	≥600 psig ≥~~533~~531°F #7	≥580 psig ≥531°F
5. Feedwater Isolation		
a. Automatic Actuation Logic Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints. ~~and Allowable Values.~~ #1	
6. Auxiliary Feedwater (3)		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	≥15% of narrow range instrument span.	≥14% of narrow range instrument span.
c. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints. ~~and Allowable Values.~~ #1	

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

2 #1 #7

PR

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUE
7. Loss of Power (Continued)		
c. 480V Load Centers (Inverse Time Relays) Degraded Voltage		
Load Center	#7	
3A	419V ±5V (60 sec ±30 sec delay)	Within ±5 volts of Setpoint
3B	426V ±5V (60 sec ± 30 sec delay)	"
3C	427V ±5V (60 sec ±30 sec delay)	"
3D	436V ±5V (60 sec ± 30 sec delay)	"
4A	427V ±5V (60 sec ± 30 sec delay)	"
4B	424V ±5V (60 sec ± 30 sec delay)	"
4C	413V ±5V (60 sec ± 30 sec delay)	"
4D	412V ±5V (60 sec ± 30 sec delay)	"
8. Engineered Safety Features Actuation System Interlocks		
a. Pressurizer Pressure	≤2000 psig	≤2010 psig
b. Low T_{avg} E	~~≥553°F~~ ≥531°F #7	≥531°F, ≤535°F
9. Control Room ~~Injection~~ Isolation E		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Trip Setpoints, ~~and Allowable Values.~~	

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TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT			TRIP SETPOINT	ALLOWABLE VALUE
9.	Control Room ~~Injection~~ Isolation (Continued) E			
	c.	High Containment Radioactivity (1) E	Particulate (R-11) ≤6.1 x 10^5 CPM Gaseous (R-12) See (2)	(Same as Trip Setpoint)
	d.	Containment Isolation Manual Phase A or Phase B	N.A.	N.A.

TABLE NOTATIONS

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

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

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

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

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

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

#5

CHANNEL 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)								E
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)	R	R	1, 2, 3**, 4 #8
c. Containment Pressure-High	N.A.	R	N.A.	M	M(1)	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. High Differential Pressure Between the Steam Line Header and any Steam Line	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3**
f. High Steam Line Flow Coincident with: Generator	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3**
Low Steam~~line~~ Pressure E	S	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3** #8
or								
Low T_{avg}	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 FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

9 #5

CHANNEL 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
2. Containment Spray								
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4
b. Containment Pressure-High-High	N.A.	R	N.A.	~~M~~ N.A.	M(1)	N.A.	N.A.	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R	N.A.	~~M~~ N.A.	M(1)	N.A.	N.A.	1, 2, 3
3. Containment Isolation								
a. Phase "A" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.							
b. Phase "B" Isolation								
1) Manual Initiation	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A	1, 2, 3, 4
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	R	R	1, 2, 3, 4

#6

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

#5

CHANNEL 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 (Continued)								
3) Containment Pressure-High-High Coincident with: Containment	N.A.	R	N.A.	~~M~~ N.A.	M(1)	N.A.	N.A.	1, 2, 3 #6
Pressure-High	N.A.	R	N.A.	~~M~~ N.A.	M(1)	N.A.	N.A.	1, 2, 3
c. Containment Ventilation Isolation								
1) Containment Isolation Manual Phase A or Phase B	N.A.	N.A.	N.A.	R	N.A.	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)	R	R	1, 2, 3, 4
3) Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements. 9.							
4) High Containment Radioactivity E	S***	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3, 4 #9
4. Steam Line Isolation								
a. Manual Initiation	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)	R	R	1, 2, 3** #8

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

#5

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

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#5

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

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TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

#59

CHANNEL 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	
8. Engineered Safety Features Actuation System Interlocks									
a. Pressurizer Pressure	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3~~*~~**	)E
b. Low T_{avg}	N.A.	R	M	N.A.	N.A.	N.A.	N.A.	1, 2, 3~~*~~**	
9. Control Room Isolation									
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	~~M(1)~~ N.A)#10	R	R	~~(3)~~ 1,2,3,4*	)E
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.								
c. High Containment Radioactivity	S***	R	M	N.A.	N.A.	N.A.	N.A.	~~(3)~~ 1,2,3,4*	)E
d. Containment Isolation Manual Phase A or Phase B	N.A.	N.A.	N.A.	R	N.A.	N.A.	N.A.	1, 2, 3, 4	

TABLE NOTATION

(1) ~~Each train shall be tested~~ at least every 62 days on a STAGGERED TEST BASIS.
(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.
* ~~(3)~~ Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment or in the spent fuel pool.)E
** Surveillance not required to enter MODE 3.)#8
*** CHANNEL CHECK may consist of a SOURCE CHECK)#9

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3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING FOR PLANT OPERATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-4.

ACTION:

a. With a radiation monitoring channel Alarm/Trip Setpoint for plant operations exceeding the value shown in Table 3.3-4, adjust the Setpoint to within the limit within 4 hours or declare the channel inoperable.

b. With one or more radiation monitoring channels for plant operations inoperable, take the ACTION shown in Table 3.3-4.

c. The provisions of Specifications 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-4

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS

FUNCTIONAL UNIT	CHANNELS TO TRIP/ALARM	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	ACTION	
1. Containment						
a. Containment Atmosphere Radioactivity-High (Particulate, ~~R-11~~ or Gaseous, ~~R-12)~~ (See Note 1.)	1	1* 2	All*	Particulate $\leq 6.1 \times 10^5$ CPM Gaseous See Note 2.	26 for MODES 1, 2, 3, 4 or 27 MODES 5 AND 6	E #1
b. RCS Leakage Detection Particulate Radio-activity ~~(R-11)~~ or Gaseous Radioactivity ~~(R-12)~~	N.A.	1	1, 2, 3, 4	N.A.	26	#1
2. Spent Fuel Storage Pool Areas						
a. Unit 3 Radioactivity - High Gaseous ~~(Rad-6418 Channel 5)~~ 2	1	1	**	$<5.5 \times 10^{-2}$ $\frac{\mu Ci}{cc}$	28	#1
b. Unit 4 Radioactivity-High Gaseous# ~~(Rad-6304 Channel 5~~ 2 ~~or R-14)~~ 2 (Plant vent) E	1	1	**	$<2.8 \times 10^{-2}$ $\frac{\mu Ci}{cc}$ ~~(RAD-6304, Channel 5)~~ (SPING) or $<1.0 \times 10^6$ CPM ~~(R-14)~~ (PRMS)	28	#1

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TABLE 3.3-4 (Continued)

TABLE NOTATIONS

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

** With irradiated fuel in the spent fuel pits.

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

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

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

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

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

ACTION STATEMENTS

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

1) A Containment sump level monitoring system per Specification 3.4.6.1.b is OPERABLE, E

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

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

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

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

PR

TABLE 3.3-4 (Continued)

ACTION STATEMENTS (Continued)

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

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

2) Monitor containment atmosphere with area radiation monitors.

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

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

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

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION FOR PLANT OPERATIONS SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
1. Containment					
a. Containment Atmosphere Radioactivity-High ~~(R-11 and R-12)~~	S##	R	M	All	)#1
b. RCS Leakage Detection					#2
1) Particulate Radioactivity ~~(R-11)~~	S##	R	M	1, 2, 3, 4	)#1
2) Gaseous Radioactivity ~~(R-12)~~	S##	R	M	1, 2, 3, 4	
2. Spent Fuel Pool Areas					
a. Unit 3 Radioactivity -High Gaseous ~~(Rad-6418 Channel 5)~~ e	S##	R	M	*	)#1
b. Unit 4 Radioactivity -High Gaseous# ~~(Rad-6304 Channel 5~~ e ~~and R-14)~~ e (SPING and PRMS) (Plant Vent))E	S##	R	M	*	)#1

TABLE NOTATIONS

* With irradiated fuel in the fuel storage pool areas.

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

\## CHANNEL CHECK may consist of a SOURCE CHECK)#2

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INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The Movable Incore Detection System shall be OPERABLE with:

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

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

c. Sufficient movable detectors, drive, and readout equipment to map the above required thimbles.

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

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

b. Monitoring the QUADRANT POWER TILT RATIO*, or

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. Monitoring the QUADRANT POWER TILT RATIO,* or

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

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

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INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-5.

ACTION:

a. As shown in Table 3.3-5

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-5

ACCIDENT MONITORING INSTRUMENTATION

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

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TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLI-CABLE MODES	ACTIONS	
14.	In Core Thermocouples (Core Exit Thermo-couples)	4/core quadrant	2/core	1, 2, 3	31, 32	
15.	Containment High Range Area Radiation ~~(Rad-6311 A and B)~~	2	1	1, 2, 3	34	#8
16.	Reactor Vessel Level Monitoring System	2(1)	1(1)	1, 2, 3	37, 38	
17.	Neutron Flux, Backup NIS (Wide Range)	2	1	1, 2, 3	31, 32	
~~18.~~	~~Containment Hydrogen Monitors~~	~~2~~	~~1~~	~~1, 2~~	~~35~~	#1
18 ~~19.~~	High Range-Noble Gas Effluent Monitors					
	a. Plant Vent Exhaust ~~(Rad-6304 Channel 9)~~	1	1	ALL	34	#8
	b. Unit 3-Spent Fuel Pit Exhaust ~~(Rad-6418 Channel 9)~~	1	1	ALL	34	
	c. Condenser Air Ejectors ~~(Rad-6417 Channel 9)~~	1	1	1, 2, 3	34	
	d. Main Steam Lines ~~(Rad-6426)~~	1	1	1, 2, 3	34	
19 ~~20.~~	RWST Water Level	2	1	1, 2, 3	31, 32	

TABLE NOTATIONS

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

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TABLE 3.3-5 (Continued)

ACTION STATEMENTS

ACTION 31 With the number of OPERABLE accident monitoring instrumentation channel(s) less than the Total Number of Channels either restore the inoperable channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

ACTION 32 With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE, either restore the inoperable channel(s) to OPERABLE status within 48 hours, or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours.

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

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

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

2) Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within ~~14~~ 30 days following the event outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to OPERABLE status. #9

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

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

ACTION 37 With the number of OPERABLE channels one less than the Total Number of Channels, restore the system to OPERABLE status within 7 days. If repairs are not feasible without shutting down, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

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TABLE 3.3-5 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

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

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

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

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

TABLE 4.3-4

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	
1. Containment Pressure (Wide Range)	M(~~3~~)	R	N.A.	#2
2. Containment Pressure (Narrow Range)	M(~~3~~)	R	N.A.	
3. Reactor Coolant Outlet Temperature - T_{HOT} (Wide Range)	M	R	N.A.	
4. Reactor Coolant Inlet Temperature - T_{COLD} (Wide Range)	M	R	N.A.	
5. Reactor Coolant Pressure - Wide Range	M	R	N.A.	
6. Pressurizer Water Level	S	R	M	
7. Auxiliary Feedwater Flow Rate	M	R	N.A.	
8. Reactor Coolant System Subcooling Margin Monitor	M	R	N.A.	
9. PORV Position Indicator (Primary Detector)	M	~~N.A.~~ R	R	#:
10. PORV Block Valve Position Indicator	M	~~N.A.~~ R	R	
11. Safety Valve Position Indicator (Primary Detector)	M(2)	R	N.A.	#4
12. Containment Water Level (Narrow Range)	M(~~3~~)	R	N.A.	
13. Containment Water Level (Wide Range)	M(~~3~~)	R	N.A.	
14. In Core Thermocouples (Core Exit Thermocouples)	M(~~3~~)	R	N.A.	#7
15. Containment - High Range Area Radiation Monitor	S(~~3~~)	R(~~2~~)'	M(3)	
16. Reactor Vessel Level Monitoring System	M(~~3~~)	R	N.A.	

P/2

TABLE 4.3-4 (Continued)

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	
17. Neutron Flux, Backup NIS (Wide Range)	M	R(4)	R(6)	)#5
~~18. Containment Hydrogen Monitor~~	~~S~~	~~Q(2)~~	~~M~~	)#1 #2
~~19.~~ 18 High Range - Noble Gas Effluent Monitors				
a. Plant Vent Exhaust ~~(Rad-6304 Channel 9)~~	~~S(5)~~ M(3)	R	M	#8
b. Unit 3 - Spent Fuel Pit Exhaust ~~(Rad-6418 Channel 9)~~	~~S(5)~~ M(3)	R	M	)#6
c. Condenser Air Ejectors ~~(Rad-6417 Channel 9)~~	~~S(5)~~ M(3)	R	M	
d. Main Steam Lines ~~(Rad-6426)~~	~~S(5)~~ M(3)	R	M	
~~20.~~ 19 RWST Water Level	M~~(1)~~	R	R	)#7

2

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TABLE 4.3-4 (Continued)

TABLE NOTATIONS

1. These are not analog channels. The requirement is satisfied if the channel test includes the verification and necessary adjustment of the channel such that the channel output results in the proper indications and/or alarm functions. #

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

3. In addition to requirements in MODES 1, 2 and 3, the surveillance for these channels shall be performed within one surveillance interval prior to heatup above 200°F. #7

2 4. By observation of the acoustic monitor power light "ON" and test of the associated alarm.

3 5. ~~By observation of related parameters~~ #6
SOURCE CHECK

4 6. Neutron detectors may be excluded from CHANNEL CALIBRATION.

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INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-6 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 24 ~~8~~ hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5). ~~If the inoperable zone(s) are not restored to OPERABLE status within 14 days, prepare and submit a Special Report to the Commission.~~)#2)#1

b. With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 3.3-6 inoperable, or with any Function B fire detection instruments shown in Table 3.3-6 inoperable, or with any two or more adjacent fire detection instruments shown in Table 3.3-11 inoperable, within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 24 ~~8~~ hours (or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5).)#2

~~c. With the fire watch patrol not established at the 18 foot level of the turbine area, restore the fire watch patrol within one hour, or prepare and submit a Special Report to the Commission.~~)#3

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

SURVEILLANCE REQUIREMENTS

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

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

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TABLE 3.3-6
FIRE DETECTION INSTRUMENTS FOR ESSENTIAL EQUIPMENT

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

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TABLE 3.3-6 (Continued)

FIRE DETECTION INSTRUMENTATION
FOR ESSENTIAL EQUIPMENT

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

TABLE NOTATIONS

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

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

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

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

#: ~~A fire watch patrol shall be established to inspect the 18 foot level of the Turbine Area once each hour.~~ #3

INSTRUMENTATION

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times, except as indicated in Table 3.3-7.) #3

ACTION:

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

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-7

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION	
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release			
a. Liquid Radwaste Effluent Line ~~(R-18)~~	1 *	35	#1
b. Steam Generator Blowdown Effluent Line ~~(R-19)~~	1 **	36	
2. Flow Rate Measurement Devices			#3
a. Liquid Radwaste Effluent Line	1 *	37	
b. Steam Generator Blowdown Effluent Lines	1/steam generator **	37	

* Applicable during liquid effluent releases.
** Applicable during blowdown operations.
#3

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TABLE 3.3-7 (Continued)

ACTION STATEMENTS

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

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

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

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

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

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

TABLE 4.3-5

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT		CHANNEL CHECK	~~SOURCE CHECK~~	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	
1. Gross Radioactivity Monitors Providing Alarm and Automatic Termination of Release						
a. Liquid Radwaste Effluent Line ~~(R-18)~~	#1	D(4), P(4)	~~P~~	R(2)*	Q(1)	#2
b. Steam Generator Blowdown Effluent Line ~~(R-19)~~		D(4)	~~M~~	R(2)	Q(1)	
2. Flow Rate Measurement Devices						
a. Liquid Radwaste Effluent Line		D(3)	~~N.A.~~	R*	Q	
b. Steam Generator Blowdown Effluent Line		D(3)	~~N.A.~~	R	Q	

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

TABLE NOTATIONS

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

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

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

(4) CHANNEL CHECK may consist of a SOURCE CHECK.

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RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-8

ACTION:

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

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated OPERABLE by performance of the CHANNEL CHECK, SOURCE CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-7.

TABLE 3.3-8

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION	
1. GAS DECAY TANK SYSTEM				
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor, ~~R-14~~)	1	*	45	#5
b. Effluent System Flow Rate Measuring Device ~~(Rad-6304 Flow Indicator Channel 10)~~	1	*	46	
2. GAS DECAY TANK SYSTEM (Explosive Gas Monitoring System)				
a. Hydrogen and Oxygen Monitors	1	**	49	
3. Condenser Air Ejector Vent System				
a. Noble Gas Activity Monitor ~~(R-15 or Rad-6417 Channel 5)~~ (SPING or PRMS	1	#	47	#5
~~b. Iodine Sampler (Rad-6417 Channel 3)~~	~~1~~	~~##~~	~~48~~	#1
~~c. Particulate Sampler (Rad-6417 Channel 1)~~	~~1~~	~~##~~	48	
b. ~~d.~~ Effluent System Flow Rate Measuring Device	1	##	46	
c. ~~e.~~ Sampler Flow Rate Measuring Device ~~(Rad-6417 Flow Indicator)~~	1	##	46	#5

PR

TABLE 3.3-8 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION	
4. Plant Vent System (Includes Unit 4's Spent Fuel Pool)				
a. Noble Gas Activity Monitor ~~(R-14 or Rad-6304 Channel 5)~~ (SPING or PRMS)	1	*	47	#5
~~b. Iodine Sampler (Rad-6304 Channel 3)~~	~~1~~	~~*~~	~~48~~	#1
~~c. Particulate Sampler (Rad-6304 Channel 1)~~	~~1~~	~~*~~	~~48~~	#1
b. ~~d.~~ Effluent System Flow Rate Measuring Device. ~~(Rad-6304 Flow Indicator Channel 10)~~	1	*	46	#5
c. ~~e.~~ Sampler Flow Rate Measuring Device ~~(Rad-6304 Flow Indicator)~~	1	*	46	#5
5. Unit 3 Spent Fuel Pit Building Vent				
a. Noble Gas Activity Monitor ~~(Rad-6418 Channel 5)~~	1	*	47	#5
~~b. Iodine Sampler (Rad-6418 Channel 3)~~	~~1~~	~~*~~	~~48~~	#1
~~c. Particulate Sampler (Rad-6418 Channel 1)~~	~~1~~	~~*~~	~~48~~	#1
b ~~d.~~ Sampler Flow Rate Measuring Device ~~(Rad-6418 Flow Indicator)~~	1	*	46	#5

P2

PR

TABLE 3.3-8 (Continued)

TABLE NOTATIONS

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

ACTION STATEMENTS

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

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

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

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

~~ACTION 48 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2 and analyzed at least weekly.~~ 2 #1

ACTION 49 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of the GAS DECAY TANK SYSTEM may continue ~~for up to 30 days~~ provided that grab samples are collected and analyzed for hydrogen and oxygen concentration at least a) once per 4 hours during degassing operations, and b) once per day during other operations. E =4

8

TABLE 4.3-6

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

2

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. GAS DECAY TANK SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (Plant Vent Monitor, ~~R-14~~)	D(4), P(7)	P	R(3)	Q(1)	* #3 #2
b. Effluent System Flow Rate Measuring Device ~~(Rad-6304 Flow Indicator Channel 10)~~ #5	P	N.A.	R	N.A.	*
2. GAS DECAY TANK SYSTEM (Explosive Gas Monitoring System)					
a. Hydrogen and Oxygen Monitors	D	N.A.	Q(4,5)	M	**
3. Condenser Air Ejector Vent System					
a. Noble Gas Activity Monitor ~~(R-15 and Rad-6417 Channel 5)~~ (SPING and PRMS)	D(7)	M	R(3)	Q(2)	# #2 #5
~~b. Iodine Sampler (Rad-6417 Channel 3)~~ 2	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~##~~ 2 #1
~~c. Particulate Sampler (Rad-6417 Channel 1)~~ 2	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~##~~ 2
6. ~~d.~~ Effluent System Flow Rate Measuring Device	D	N.A.	R	N.A.	##

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TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	~~SOURCE CHECK~~ 2	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
3. Condenser Air Ejector Vent System (Continued)						
c ~~e~~. Sampler Flow Rate Measuring Device ~~(Rad-6417 Flow Indicator)~~ #5	D	~~N.A.~~	R	N.A.	##	
4. Plant Vent System (Includes Unit 4's Spent Fuel Pool)						
a. Noble Gas Activity Monitor ~~(R-14 and Rad-6304 Channel 5)~~ (SPING and PRMS) #5	D(7)	~~M~~	(3,6)	Q(2)	*	#2
~~b. Iodine Sampler (Rad-6304 Channel 3)~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~ 2	#1
~~c. Particulate Sampler (Rad-6304 Channel 1)~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~ 2	#1
b ~~d~~. Effluent System Flow Rate Measuring Device ~~(Rad-6304 Flow Indicator Channel 10)~~ #5	D	~~N.A.~~	(6)	N.A.	*	
c ~~e~~. Sampler Flow Rate Measuring Device ~~(Rad-6304 Flow Indicator)~~	D	~~N.A.~~	(6)	N.A.	*	

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TABLE 4.3-6 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	~~SOURCE CHECK~~	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED	
5. Unit 3 Spent Fuel Pit Building Vent						
a. Noble Gas Activity Monitor ~~(Rad-6418 Channel 5)~~ #5	D(7)	~~M~~	R(3)	Q(2)	*	#2
~~b. Iodine Sampler (Rad-6418 Channel 3)~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~	#1
~~c. Particulate Sampler (Rad-6418 Channel 1)~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~	#1
b ~~d.~~ Sampler Flow Rate Measuring Device ~~(Rad-6418 Flow Indicator)~~ #5	D	~~N.A.~~	R	N.A.	*	

TABLE NOTATIONS

* At all times.

** During GAS DECAY TANK SYSTEM operation.

\# Applies during MODES 1, 2, 3 and 4

\#\# Applies during MODES 1, 2, 3 and 4 when primary to secondary leakage is detected as indicated by condenser air ejector noble gas activity monitor

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

(2) The ANALOG CHANNEL OPERATIONAL TEST shall also demonstrate that control room or computer room alarm annunciation ~~(for Rad-6304, Rad-6417 and Rad-6418 alarm annunciation is in the computer room)~~ occurs if the instrument indicates measured levels above the Alarm Setpoint. E

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

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TABLE 4.3-6 (Continued)

TABLE NOTATIONS (Continued)

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

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

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

a. One volume percent oxygen, balance nitrogen, and

b. Four volume percent oxygen, balance nitrogen.

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

(7) CHANNEL CHECK may consist of a SOURCE CHECK. #2

PR

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 6 hours.

SURVEILLANCE REQUIREMENTS

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

PR

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 All of the reactor coolant loops listed below shall be OPERABLE with all reactor coolant loops in operation when the Reactor Trip ~~System~~ breakers are closed and two reactor coolant loops listed below shall be OPERABLE with at least one reactor coolant loop in operation when the Reactor Trip ~~System~~ breakers are open~~*~~: *, [**] E #1

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

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

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

APPLICABILITY: MODE 3.

ACTION:

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

b. With less than three reactor coolant loop in operation and the Reactor Trip ~~System~~ breakers in the closed position, within 1 hour open the Reactor Trip ~~System~~ breakers.

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.2.2 The required steam generators shall be determined OPERABLE by verifying secondary side water level to be greater than or equal to 10% (narrow range) at least once per 12 hours. E

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

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

** As an alternative to the reactor trip breakers open, the rod control system may be placed in the bank select mode with a shutdown bank selected.

PR

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loops listed below shall be OPERABLE and at least one of these loops shall be in operation:*

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

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

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

d. RHR Loop A (***) ~~, and~~

e. RHR Loop B (***)

#1

APPLICABILITY: MODE 4.

ACTION:

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

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

E

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

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

*** An inoperable RHR loop may be the operating RHR loop for up to 2 hours for surveillance testing to establish operability.

#1

PR

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

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

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

narrow range E

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

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 10%.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

---

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

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

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

PR

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

---

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

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

PR

REACTOR COOLANT SYSTEM
3/4.4.2 Safety Valves
OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.~~2~~ 1 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%.* E

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~~ 1 No additional requirements other than those required by Specification 4.0.5. E

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

PR

REACTOR COOLANT SYSTEM

~~3/4.4.2~~ SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.~~1~~ 2 A minimum of one pressurizer Code safety valve shall be OPERABLE with a lift setting of 2485 psig ± 1%.* E

APPLICABILITY: MODE~~S~~ 4 and MODE 5~~.~~ with the RCS Pressure Boundary established.** #2

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes ~~and place an OPERABLE RHR loop into operation in the shutdown cooling mode.~~ except that an OPERABLE RHR loop shall be placed into operation in the shutdown cooling mode. #1

SURVEILLANCE REQUIREMENTS

4.4.2.~~1~~ 2 No additional requirements other than those required by Specification 4.0.5. E

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

** RCS Pressure Boundary is established unless the RCS is vented to the containment with a minimum opening of greater than or equal to 2.20 square inches. 2

E

PR

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per ~~18 months~~ by demonstrating the capability to power the heaters from the emergency power. #2

refueling, not to exceed 24 months

PR

INSERT FPL REWRITE

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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 with the motive force supplied by the normal Instrument Air System and the backup Nitrogen Gas System.

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 or is closed due to PORV leakage.

USE THIS FPL REWRITE

REACTOR COOLANT SYSTEM

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each Power Operated Relief Valve (PORV) Block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one or more block valve(s) inoperable, within 1 hour either restore the block valve(s) to OPERABLE status or close the block valve(s) and remove power from the block valve(s); otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 6 hours. #1 #2

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4 Each block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel unless the block valve is closed with power removed in order to meet the requirements of ~~Action a. in~~ Specification 3.4.4. or is closed ~~due to PORV leakage~~ when providing an isolation function.

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REACTOR COOLANT SYSTEM

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 within 1 hour ~~prior to increasing T_{avg} above 200°F.~~ or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours. #1

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 refueling, not to exceed 24 months, ~~shutdown~~ by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1. #2

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:

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SURVEILLANCE REQUIREMENTS (Continued)

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

2) Tubes in those areas where experience has indicated potential problems, and

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

c. The tubes selected as the second and third samples in the inservice inspection may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to be inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

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

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

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

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

a. The first inservice inspection shall be performed after 6 Effective Full Power Months but within 24 calendar months following replacement of steam generators. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months;

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results 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 a subsequent inspection demonstrates that a third sample inspection is not required.

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

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

   2) A seismic occurrence greater than the Operating Basis Earthquake, or

   3) A loss-of-coolant accident resulting in rapid depressurization of the primary system, or

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

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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. Any tube which does not permit the passage of the eddy current inspection probe shall be deemed a defective tube;

6) Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness;

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

8) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg, or from the point of entry (cold leg side) completely around the U-bend and to the bottom of the hot leg; and

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REACTOR COOLANT SYSTEM

STEAM GENERATOR

SURVEILLANCE REQUIREMENTS (Continued)

9) Preservice Inspection means an inspection of the full length of each tube in each steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing.

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

4.4.5.5 Reports

a. Within ~~15~~ 30 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; #3

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

1) Number and extent of tubes inspected,

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

3) Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 shall be reported to the Commission pursuant to 10 CFR Part 50.72 and prior to resumption of plant operation. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

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

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TABLE 4.4-1 MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED
DURING INSERVICE INSPECTION

STEAM GENERATOR TUBE INSPECTION

SAMPLE SIZE	1st SAMPLE INSPECTION		2nd SAMPLE INSPECTION		3rd SAMPLE INSPECTION	
	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect additional 2S tubes in this S.G.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect additional 2S tubes tubes in this S.G.	C-1	None
					C-2	Plug defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G. plug defective tubes and inspect 2S tubes in each other S.G. Notification to NRC pursuant to Paragraph 50.72(b)(2) of 10 CFR 50	All other S.G.s are C-1	None	N/A	N/A
			Some S.G.s C-2 but no additional S.G.s are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to Paragraph 50.72(b)(2) of 10 CFR 50	N/A	N/A

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

TABLE 4.4-2 STEAM GENERATOR TUBE INSPECTION

P2

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3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and

b. A Containment Sump Level Monitoring System.

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

ACTION:

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

1) A Containment Sump Level Monitoring System is OPERABLE;

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

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

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

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

one Containment Sump Level Monitoring

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection Systems shall be demonstrated OPERABLE by:

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

b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION at least once per ~~18 months.~~ refueling, not to exceed 24 months #1.

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REACTOR COOLANT SYSTEM

e. 30 gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig, and |#1

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 ~~reactor~~-to-secondary leakage through all steam generators ~~not isolated from the Reactor Coolant System~~ and 500 gallons per day through any one steam generator.) E

d. 10 GPM IDENTIFIED LEAKAGE from the Reactor Coolant System, |#1

f ~~e~~. ~~0.5 GPM~~ Leakage as specified in Table 3.4-1 ~~per nominal inch of valve size~~ up to a maximum of 5 GPM at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.* |#2

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

ACTION:

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

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

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

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

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

PR

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION (Continued)

2. The leakage* from the remaining isolation valves in each high pressure line having a valve not meeting the criteria of Table 3.4-1, as listed in Table 3.4-1, shall be determined and recorded daily. The positions of the other valves located in the high pressure line having the leaking valve shall be recorded daily unless they are manual valves located inside containment.)#3

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

d. With any Reactor Coolant System Pressure Isolation Valve leakage greater than 5 gpm, reduce leakage to below 5 gpm ~~, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~ within 1 hour)#4

SURVEILLANCE REQUIREMENTS

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

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

b. ~~Monitoring the containment sump level at least once per 12 hours;~~)#5

INSERT A

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

e ~~d~~. Monitoring the Reactor Head Flange Leakoff System at least once per 24 hours.)E

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage* to be within its limit:)#3

INSERT C

a. At least once per refueling ~~shutdown~~ not to exceed 24 months)E

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

c. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could potentially result in leakage.)#6

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

INSERT B)#3

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

INSERT A To T.S. 3/4.4.6.2

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days. The provisions of Specification 4.0.4 are not applicable for entry into Mode 3 or 4; #5

INSERT B To T.S. 3/4.4.6.2

* To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. #3

INSERT C To T.S. 3/4.4.6.2

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

1. Within 24 hours by verifying valve closure, and

2. Prior to entering Mode 2 by verifying leakage rate, #9

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TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

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

ACCEPTABLE LEAKAGE LIMITS

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

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

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

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

PR

REACTOR COOLANT SYSTEM

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3, and 4:

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

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*	< 0.10 ppm	≤ 1.00 ppm
Chloride **	< 0.15 ppm	≤ 1.50 ppm
Fluoride **	≤ 0.15 ppm	≤ 1.50 ppm

#1

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

#1

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

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

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

#1

#1

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

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

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3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. Less than or equal to 1 microCurie per gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microCuries per gram of gross radioactivity.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with average reactor coolant temperature less than 500°F within 6 hours; and

b. With the specific activity of the reactor coolant greater than 100/$\bar{E}$ microCuries per gram, be in at least HOT STANDBY with average reactor coolant temperature less than 500°F within 6 hours.)E

And in

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

With the specific activity of the reactor coolant greater than 1 microCurie per gram DOSE EQUIVALENT I-131 or greater than 100/$\bar{E}$ microCuries per gram, perform the sampling and analysis requirements of Item 6.a) of Table 4.4-4 until the specific activity of the reactor coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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

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

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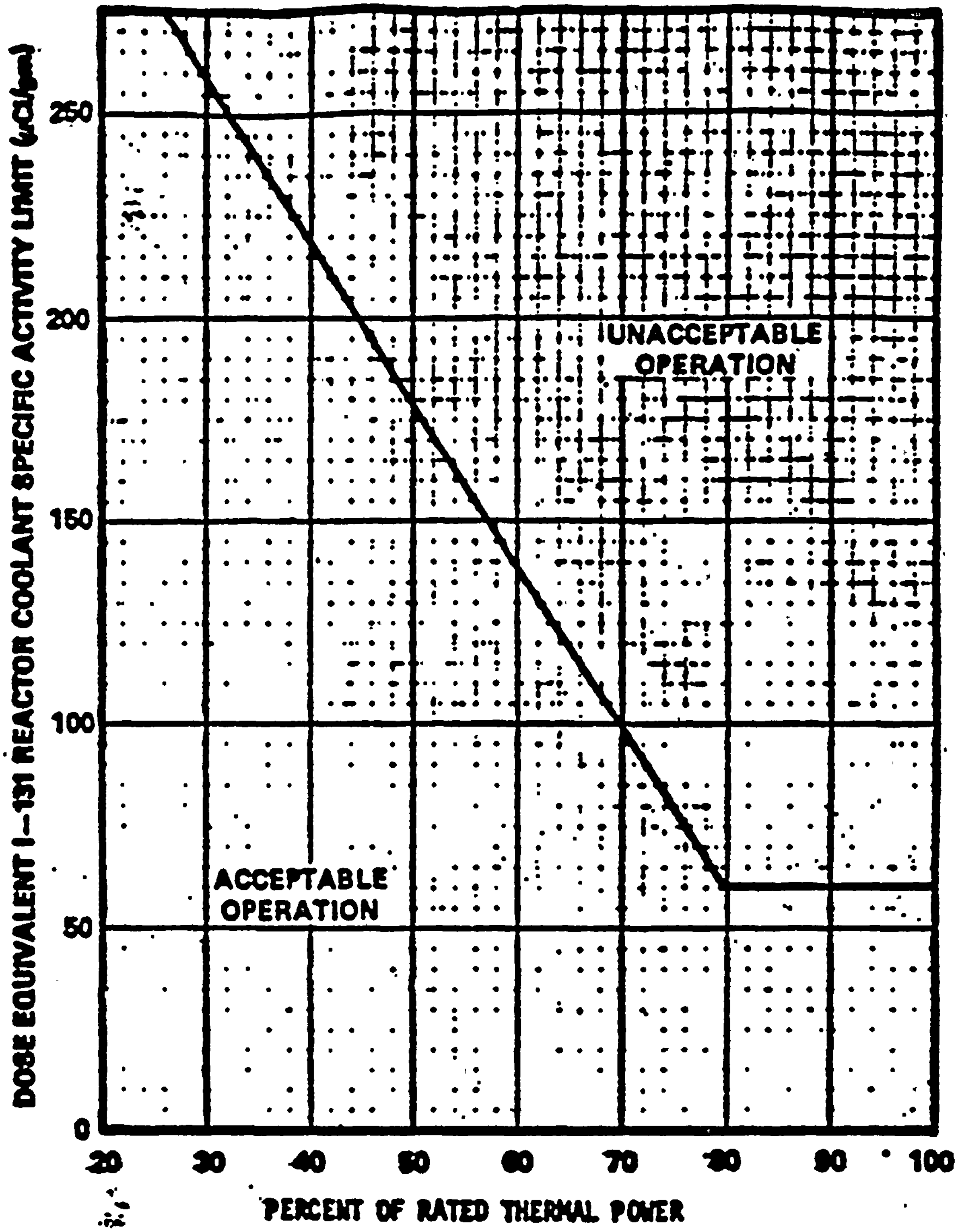

FIGURE 3.4-1

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

TABLE 4.4-4

REACTOR COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Radioactivity Determination	At least once per 72 hours.	1, 2, 3, 4
2. Tritium Activity Determination	1 per 7 days.	1, 2, 3, 4
3. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days.	1
4. Radiochemical Isotopic Determination Including Gaseous Activity	Monthly	1, 2, 3, 4
5. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
6. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 1 µCi/gram DOSE EQUIVALENT I-131 or 100/$\bar{E}$ µCi/gram of gross radioactivity, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15% of the RATED THERMAL POWER within a 1-hour period.	1, 2, 3

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TABLE 4.4-4 (Continued)

TABLE NOTATIONS

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

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

PR

REACTOR COOLANT SYSTEM

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 for Unit 3 and Figures 3.4-4 and 3.4-5 for Unit 4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

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

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

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

APPLICABILITY: At all times.

ACTION:

ADD ATTACHED INSERT

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.

#1

SURVEILLANCE REQUIREMENTS

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

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

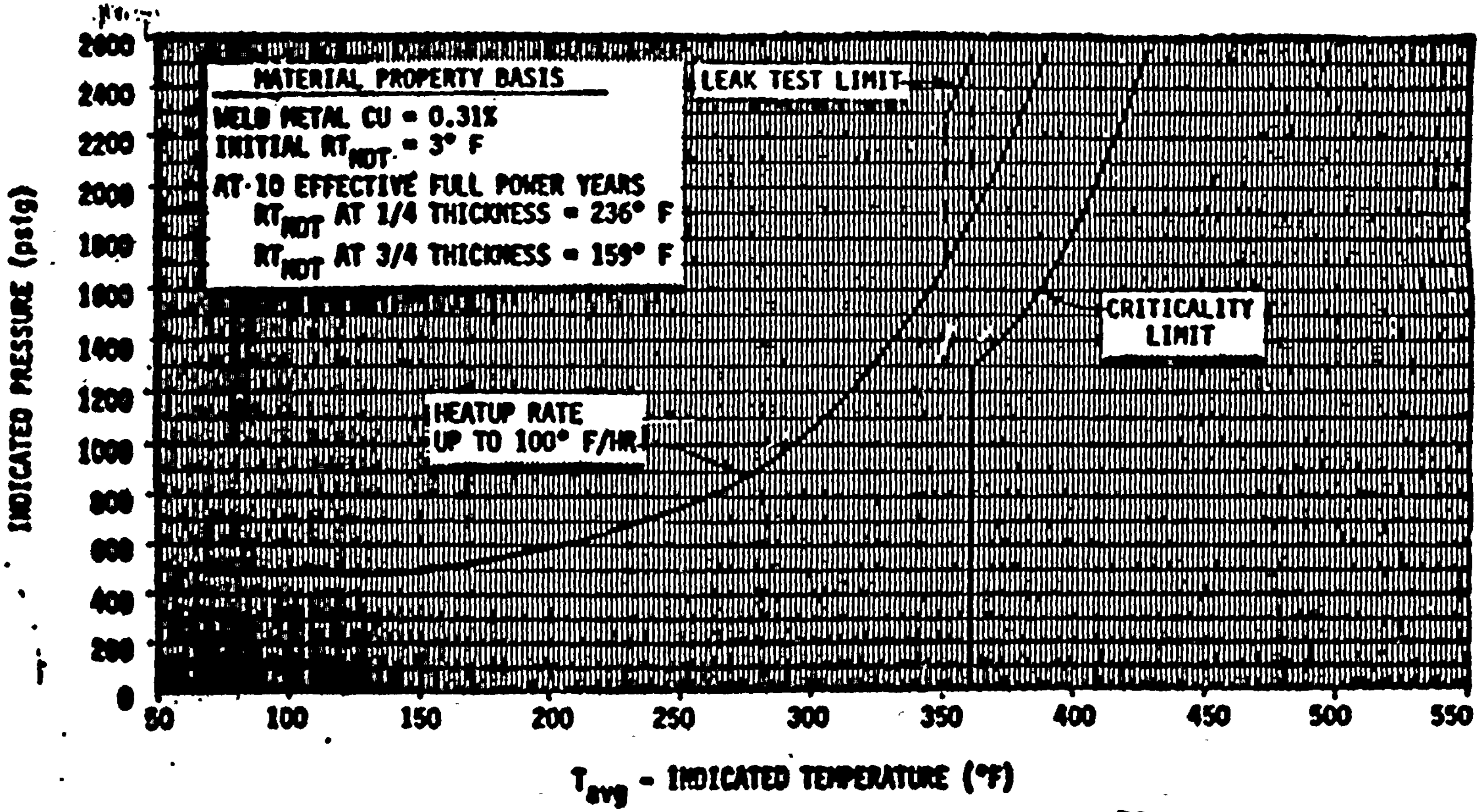

FIGURE 3.4-2
TURKEY POINT UNIT 3
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 10 EFPY

INSERT FOR 3.4.9.1 ACTION

With any of the above limits exceeded:

1. Restore the temperature and/or pressure to within the limit within 30 minutes, and
2. Within 6 hours either:
   a. Complete an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the RCS and determine that the RCS remains acceptable for continued operation, or
   b. Be in HOT STANDBY and reduce the average coolant temperature and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours, and complete the engineering evaluation in 2.a above prior to exceeding either 200°F or 500 psig.

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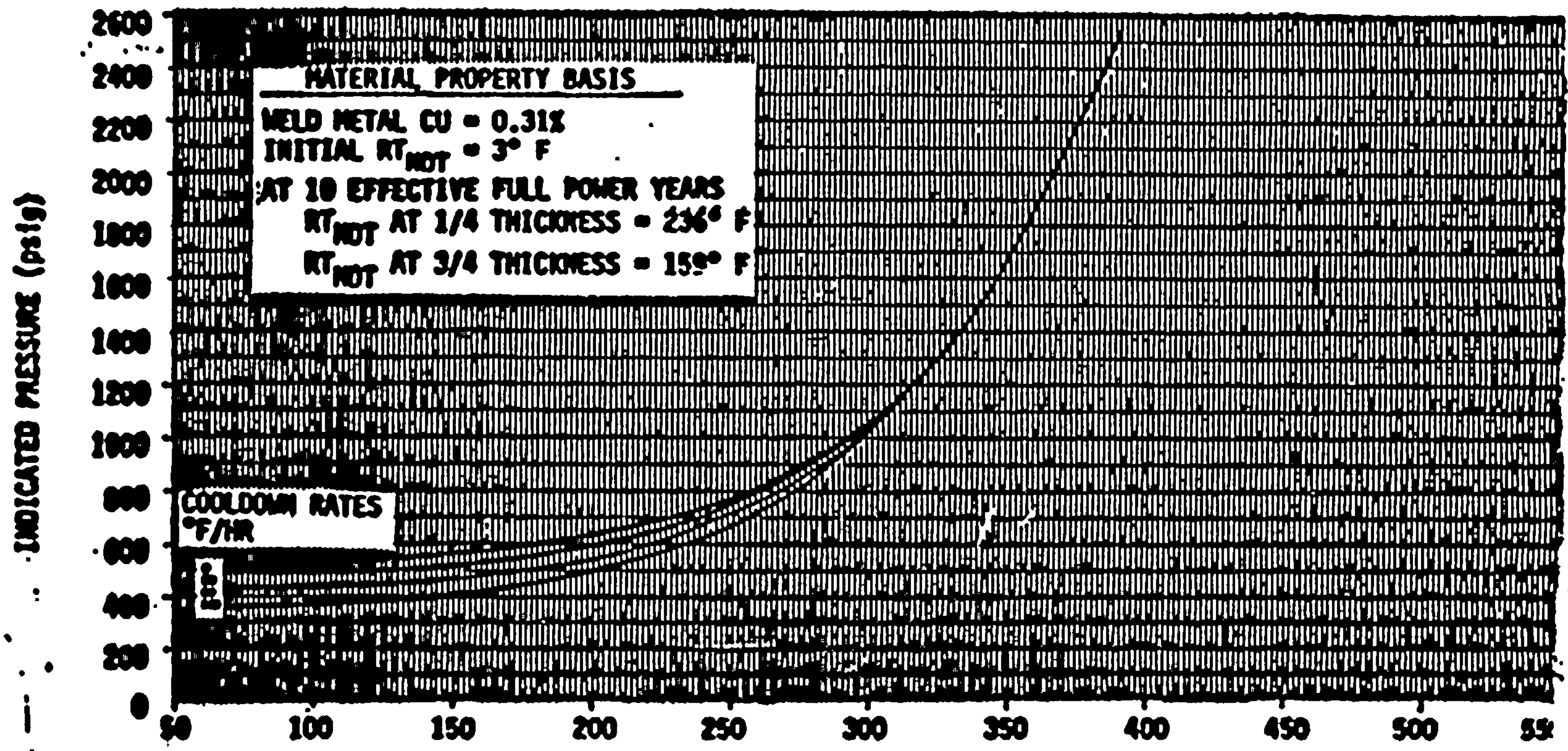

RCS Average Temperature ~~T_{avg}~~ - INDICATED TEMPERATURE (°F)

FIGURE 3.4-3
TURKEY POINT UNIT 3
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 10 EFPY

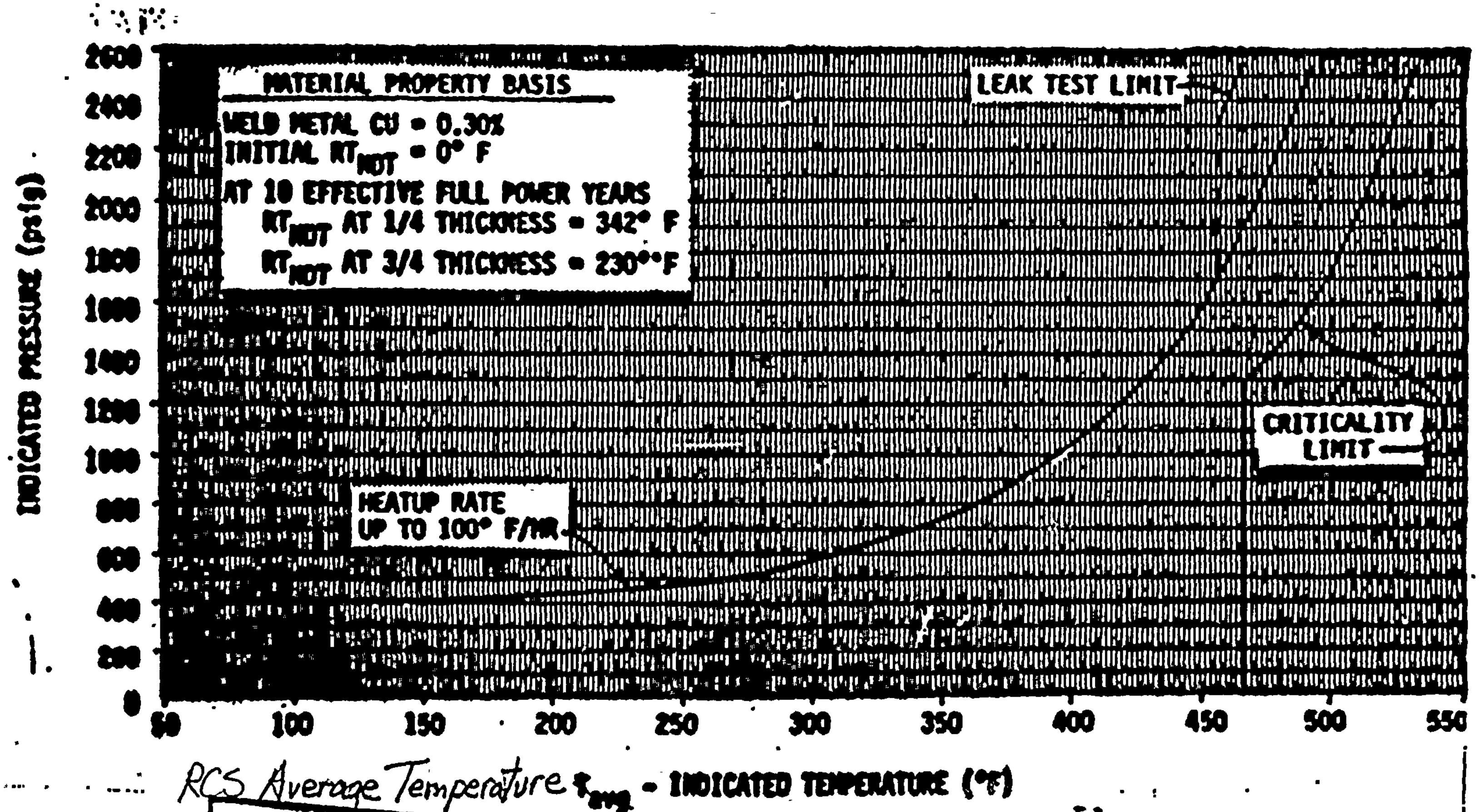

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FIGURE 3.4-4
TURKEY POINT UNIT 4
REACTOR COOLANT SYSTEM HEATUP LIMITATIONS - APPLICABLE UP TO 10 EFPY

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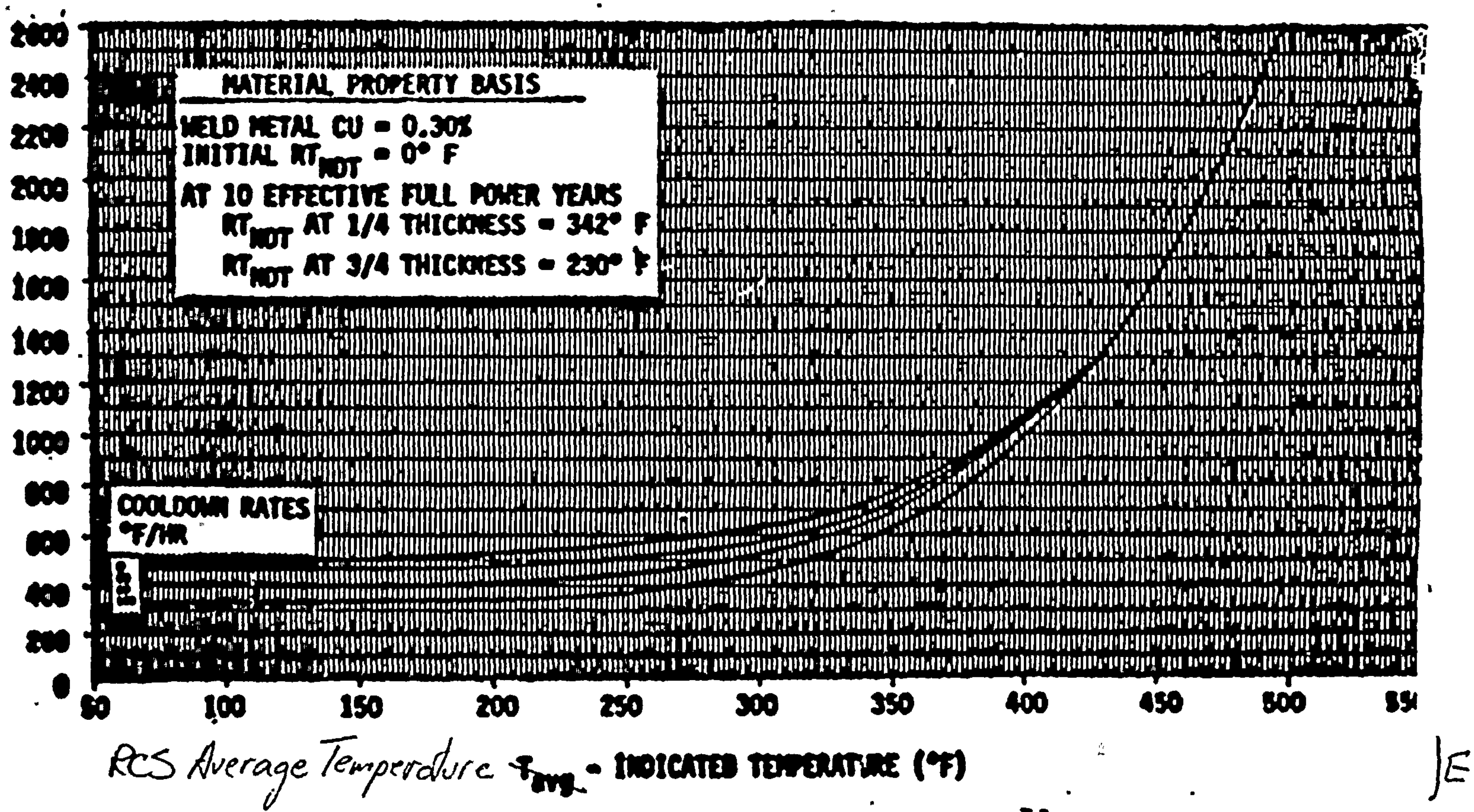

FIGURE 3.4-5
TURKEY POINT UNIT 4
REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS - APPLICABLE UP TO 10 EFPY

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TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

UNIT 3

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
U	30°	0.49	Standby
~~V~~			~~12 years~~
W	40°	0.34	Standby
X	50°	0.34	33 years
Y	150°	0.49	Standby
Z	230°	0.34	Standby

UNIT 4

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

#2

PR

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING ~~PROTECTION~~ SYSTEMS)E

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODE 3*, 4, 5 and 6 with the reactor vessel head ~~on~~ not removed) |E

ACTION:

a. With the high pressure safety injection capability to the RCS unisolated, take immediate action to isolate this capability.

b. In MODE 4 with RCS average coolant temperature less than or equal to 275°F, and in MODE 5 or in MODE 6 with the reactor vessel ~~head on:~~ not removed:

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

a. ~~1.~~ Restore the inoperable PORV to OPERABLE status, or

b. ~~2.~~ Depressurize and vent the RCS through at least a 2.20 square inch, or

c. ~~3.~~ Depressurize and maintain a RCS vent through at least one open PORV and open associated block valve.

E

~~*When RCS average coolant temperature is less than or equal to 380°F.~~

*During cool down, high head safety injection capability to the RCS shall be isolated after reducing RCS cold leg temperature to ≤ 380°F and prior to reaching 350°F.)#1

PR

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F 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 ~~300°F~~ 320 F. #1

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. #2

ADD ATTACHED INSERT

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.

INSERT FOR 3.4.9.2 ACTION

With the pressurizer temperature limits in excess of the above limits:

1. Restore the temperature to within the limits within 30 minutes, and
2. Within 6 hours either:
   a. Complete an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer and determine that the pressurizer remains acceptable for continued operation, or
   b. Be in HOT STANDBY and reduce the pressurizer pressure to less than 500 psig within the following 30 hours and complete the engineering evaluation in 2.a above prior to exceeding 500 psig.

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REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

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

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once each refueling~~; and~~ not to exceed 24 months

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

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

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

4.4.9.3.3 Verify the high pressure injection capability to the RCS is isolated at least once per 24 hours.

*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.

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REACTOR COOLANT SYSTEM

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.

SURVEILLANCE REQUIREMENTS

4.4.10 Requirements of Specification 4.0.5 shall be met including the following specific requirements.

a. Reactor Coolant System integrity shall be demonstrated as follows after the system is closed following normal opening, modification or repair.

1) When the Reactor Coolant System is closed, the system will be leak tested at not less than 2335 psig while meeting NDTT requirements for temperature.

2) When Reactor Coolant System modifications or repairs have been made which involved new strength welds on components greater than ~~2 in.~~ diameter, the new welds will receive both a surface and 100% volumetric examination. #2

or equal to 4 in.

PR

SURVEILLANCE REQUIREMENTS (Continued)

3) When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components ~~2 in.~~ diameter ~~or smaller~~, the new welds will receive a surface examination. #2

less than 4 in.

PR

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

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

a. Reactor vessel head, and

b. Pressurizer steam space

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.11 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per ~~18 months~~ refueling, not to exceed 24 months by: #1

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

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

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

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PR

EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open and its circuit breaker open,

b. A contained borated water volume of between 6545 and 6665 gallons,

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

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

e. A water level and pressure channel OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

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

*Pressurizer pressure above 1000 psig.

PR

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

c. At least once per 31 days:

   1) By verifying that the power to the isolation valve operator is disconnected by an ~~locked~~ open breaker when the RCS pressure is above 1000 psig, and E

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

d. At least once per refueling, not to exceed 24 months: #1

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

   2) ~~During shutdown~~, each accumulator check valve~~s~~ shall be checked for operability. E

P12

EMERGENCY CORE COOLING SYSTEMS

RCS Average Coolant Temperature

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

LIMITING CONDITION FOR OPERATION

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

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

b. Two OPERABLE RHR heat exchangers,

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

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

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

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

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

~~b. In the event the ECCS is actuated and injects water in the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.~~ | #1

b ~~c~~. With one Safety Injection pump inoperable, restore the pump to OPERABLE status within 30 days or be in at least HOT STANDBY within the next ~~6~~ 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.) E | #4

c ~~d~~. With two Safety Injection Pumps inoperable, restore one of the two inoperable pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following ~~6~~ 12 hours. This ACTION applies to both units simultaneously | #4

~~*Specification 3.5.2.a is not applicable to RCS cold leg temperature of ≤380°F due to overpressurization protection requirements that isolate High Head Safety Injection from the RCS.~~

During heatup high head safety injection capability to the RCS may only be restored after reaching 350°F. Specification 3.5.2.a is applicable with RCS cold leg temperature ≥ 380°F. #

PR

SURVEILLANCE REQUIREMENTS

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

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

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

#5

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

b. At least once per 31 days by:

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

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

3) Verifying that each RHR Pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5:

RHR Pump $\geq$ 131 psid at a metered flowrate $\geq$ 150 gpm (recirculation mode), or

$\geq$ 231 ft at metered flowrate $\geq$ 3600 gpm (normal cooldown mode).

c. At least once per 92 days by:

1) Verifying that each SI pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

e. At least once per ~~18 months~~ refueling, not to exceed 24 months by: #6

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

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

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

f. At least once per ~~18 months, during shutdown,~~ refueling, not to exceed 24 months by: #6

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

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

      a) Safety Injection pump, and

      b) RHR pump.

PR

SURVEILLANCE REQUIREMENTS

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

1) Within 4 hours following completion of each valve stroking operation or maintenance on the valve when the ECCS ~~subsystems~~ equipment ~~are~~ required to be OPERABLE, and | E

2) At least once per 18 months. | E

RHR ~~LPSI~~ System
Valve Number

~~HCV-*-758~~
MOV-*-872
~~*-887~~
FCV-*-605

| #3.

PR

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - ~~T_{avg}~~ RCS Average Coolant Temperature LESS THAN 350°F

|E

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE RHR heat exchanger,

b. One OPERABLE RHR pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

b. With either the residual heat removal heat exchanger or RHR pump inoperable, restore the components to OPERABLE status or maintain the Reactor Coolant System ~~T_{avg}~~ RCS Average Coolant Temperature less than 350°F by use of alternate heat removal methods. |E

~~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.~~ e |#1

SURVEILLANCE REQUIREMENTS

4.5.3 ~~.1~~ The ECCS components shall be demonstrated OPERABLE per the applicable requirements of Specification 4.5.2.

~~4.5.3.2 All Safety Injection pumps shall be demonstrated inoperable by verifying that the motor circuit breakers are secured in the open position 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°F.~~ |#2

PR

EMERGENCY CORE COOLING SYSTEMS

~~BORON INJECTION SYSTEM~~

~~3/4.5.4~~ REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

a. A minimum indicated ~~contained~~ borated water volume of 320,000 gallons per RWST, |E

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

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

d. A maximum solution temperature of 100°F.

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

ACTION:

With less than the required number of RWST(s) OPERABLE, restore the tank(s) to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The required RWST(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the indicated ~~contained~~ borated water volume in the tank, and |E

2) Verifying the boron concentration of the water.

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

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

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

PR

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

** Exception may be taken under Administrative Controls for opening of certain valves and airlocks necessary to perform surveillance or testing requirements.

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.**) #1

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.) #1

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, ~~except as provided in Table 3.6-1 of Specification 3.6.4.1~~; and

~~b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3; and~~ #2

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

*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.

PR

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:~~ o

~~1)~~ e Less than or equal to L_a, 0.25% by weight of the containment air per 24 hours at ~~P_a,~~ 50 psig, ~~or~~)E

~~2) Less than or equal to L_t, 0.177% by weight of the containment air per 24 hours at a reduced pressure of P_t, 25 psig.~~ e)#1

b. A combined leakage rate of less than 0.60 L_a for all penetrations and valves subject to Type B and C tests, when pressurized to 50 psig.

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

ACTION:

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

b. [insert attached page])#2

SURVEILLANCE REQUIREMENTS

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

4.6.1.2.2 ~~a.~~ Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at a pressure not less than ~~either P_a,~~ 50 psig, ~~or at P_t, 25 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;)E)#1

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b. With a combined leakage rate of greater than or equal to 0.60 La for all penetrations and valves subject to type B and C tests, when pressurized to 50 psig, reduce the leakage rate to less than 0.60 La within 1 hour or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours

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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. #3

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

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

$$[\ L_c - (L_{am} + L_o)\ ] \leq 0.25\ La \text{ or } [\ L_c - (L_{tm} + L_o)\ ] \leq 0.25\ L_t$$

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

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

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

4.6.1.2.3 ~~d.~~ Type B and C tests shall be conducted with gas at a pressure not less than P_a, 50 psig, during each reactor shutdown for refueling, but in no case, at intervals ~~no~~ greater than 24 months except for tests involving: E

a ~~1)~~ Air locks,

b ~~2)~~ Purge supply and exhaust isolation valves, and E

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

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

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

4.6.1.2.6 ~~g.~~ The provisions of Specification 4.0.2 are not applicable. E

PR

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, or during the performance of containment air lock surveillance and/or testing requirements, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to ~~0.02~~ 0.2 L_a at 50 psig. #1

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~~ outer air lock door closed; =

2. Operation may then continue until performance of the next required overall air lock leakage test provided that the following applicable action is performed at least once per 31 days:

   a) If the outer air lock door is inoperable, verify by control room indication or local observation, that the inner airlock door is closed and verify the outer air lock door is locked closed, or

   b) If the inner air lock door is inoperable, verify that the outer air lock door is locked closed; E

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

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying that the seals have not been damaged and have seated properly by vacuum testing the volume between the door seals in accordance with approved plant procedures.

b. By conducting overall air lock leakage tests at not less than 50 psig, and verifying the overall air lock leakage rate is within its limit at least once per 6 months.*

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

*The provisions of Specification 4.0.2 are not applicable.

PR

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and +3 psig.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.5 Primary containment average air temperature shall not exceed 120°F.*)#1

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.~~)#1

ADD INSERT B

SURVEILLANCE REQUIREMENTS

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

Location

a.	~~TE-6700~~	0° Azimuth	@ 58 feet elevation
b.	~~TE-6701~~	120° Azimuth	@ 58 feet elevation
c.	~~TE-6702~~	240° Azimuth	@ 58 feet elevation

With any of the above ~~temperature monitor(s) out of service use the nearest backup:~~ temperature indication at locations unavailable use the nearest of the following indicator locations:)#1

a.	~~TE-1497~~	North Wall	@ 58 feet elevation
b.	~~TE-1498~~	West Wall	@ 58 feet elevation
c.	~~TE-1499~~	C-Filter	@ 58 feet elevation

E

ADD INSERT A)#1

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CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With more than one tendon with an observed lift-off force between the predicted lower limit and 90% of the predicted lower limit or with one tendon below 90% of the predicted lower limit, restore the tendon(s) to the required level of integrity within 15 days and perform an engineering evaluation of the containment and provide a Special Report to the Commission within 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With any abnormal degradation of the structural integrity other than ACTION a, at a level below the acceptance criteria of Specification 4.6.1.6.1, 4.6.1.6.2 and 4.6.1.6.3, restore the containment to the required level of integrity within 72 hours and perform an engineering evaluation of the containment and provide a Special Report to the Commission within ~~15~~ 30 days in accordance with Specification 6.9.2 or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.)E #

SURVEILLANCE REQUIREMENTS

4.6.1.~~7~~6.1 Containment Tendons. The containment tendons' structural integrity shall be demonstrated every fifth year from the date of the initial structural integrity test. The tendons' structural integrity shall be demonstrated by: E

a. Determining that a random but representative sample of at least ~~15~~ 12 tendons (~~5~~ 3 dome, ~~5~~ 4 vertical, and ~~5~~ 5 hoop) each have an observed lift-off force within predicted limits for each. For each subsequent inspection one tendon from each group may be kept unchanged to develop a history and to correlate the observed data. If the observed lift-off force of any one tendon in the original sample population lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of this tendon should be checked for their lift-off forces. If both of these adjacent tendons are found to be within their predicted limits, all three tendons should be restored to the required level of integrity. This single deficiency may be considered unique and accceptable. #5

INSERT A To T.S. 3.6.1.5

* Operation greater than 120°F but less than or equal to 125°F may continue but shall be limited to less than or equal to 336 equivalent hours during a calender year.

Time durations with containment average air temperature above 120°F may be determined equivalent to 336 hours at 125°F using the time-temperature relationships that support the environmental qualification requirements of 10 CFR 50.49.

INSERT B To T.S. 3.6.1.5

With the containment average air temperature greater than 125°F or greater than 120°F for more than 336 equivalent hours* during a calender year, reduce the average air temperature to within the applicable limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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SURVEILLANCE REQUIREMENTS (Continued)

b. Performing tendon detensioning, inspections, and material tests on a previously stressed tendon from each group (dome, vertical, and hoop). A randomly selected tendon from each group shall be completely detensioned in order to identify broken or damaged wires and determining that over the entire length of the removed wire or strand that:

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

2) There are no changes in the presence or physical appearance of the sheathing filler-grease, and

3) A minimum tensile strength of 240,000 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

c. Performing tendon retensioning of those tendons detensioned for inspection to their observed lift-off force with a tolerance limit of +6,-0% ~~±6%~~. During retensioning of these tendons, the changes in load and elongation should be measured simultaneously at a minimum of three approximately equally spaced levels of force between zero and the seating force. If the elongation corresponding to a specific load differs by more than 5% from that recorded during installation, an investigation should be made to ensure that the difference is not related to wire failures or slip of wires in anchorages; /E

d. Insert A ~~Assuring the observed lift-off stresses exceed the average minimum design value given below, which are adjusted to account for elastic losses; and~~ #6

~~Dome~~	~~133 ksi~~
~~Vertical~~	~~133 ksi~~
~~Hoop~~	~~133 ksi~~

e. Verifying the OPERABILITY of the sheathing filler grease by:

1) Minimum grease coverage exists for the different parts of the anchorage system, and

2) The chemical properties of the filler material are within the tolerance limits as specified by the manufacturer.

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SURVEILLANCE REQUIREMENTS (Continued)

— INSERT B —

~~4.6.1.6.2~~ End Anchorages and Adjacent Concrete Surfaces. The ~~structural~~ integrity of the ~~end anchorages~~ of all tendons ~~inspected pursuant~~ to Specification 4.6.1.7.1 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.~~ #2

~~4.6.1.6.3~~ 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~~. This first inspection performed will form the ~~baseline for future~~ surveillances.

4.6.1.6.3 Containment Surfaces

In accordance with 10 CFR 50, Appendix J. Section V. A, a visual inspection of the accessible interior and exterior surfaces of the containment, including the liner plate, shall be performed during the shutdown for (but prior to) each Type A containment leakage rate test (Technical Specification 4.6.1.2.1). The purpose of this inspection shall be to identify any evidence of structural deterioration which may affect containment structural integrity or leaktightness. The visual inspection shall be general in nature; its intent shall be to detect gross areas of widespread cracking, spalling, gouging, rust, weld degradation, or grease leakage. The visual examination may include the utilization of binoculars or other optical devices. Corrective actions taken, and recording of structural deterioration and corrective actions, shall be in accordance with 10 CFR 50, Appendix J, Section V. A. Records of previous inspections shall be reviewed to verify no apparent changes in appearance. The first inspection performed will form the baseline for future surveillances. #3

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INSERT A TO 3.6.1.6.

d. Assuring that the observed lift-off force for each tendon exceeds the minimum required lift-off force. Required lift-off forces shall be calculated individually for each surveillance tendon prior to the beginning of each surveillance, and should consider such factors as:

1) Prestressing history;

2) Friction losses; and

3) Time-dependent losses (creep, shrinkage, relaxation), considering time elapsed from prestressing.

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4.6.1.6.2 End Anchorages and Adjacent Concrete Surfaces

The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.6.1 and the adjacent concrete surfaces shall be demonstrated by determining through visual inspection that no unacceptable levels of corrosion exist on the end anchorages and no unacceptable cracking exists in the concrete adjacent to the end anchorages. Determination of acceptance levels shall be by engineering evaluation of the areas in question. If unacceptable conditions are found, the tendons inspected during the previous surveillance shall be examined to determine whether the corrosion levels or concrete cracking have increased since the previous surveillance. Inspection of adjacent concrete surfaces shall be performed concurrently with the containment tendon surveillance (Technical Specification 4.6.1.6.1).

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CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

REPLACE WITH ATTACHED INSERT

#1

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply (48-inch) and exhaust isolation (54-inch) valve shall be OPERABLE and shall be closed and sealed closed. Operation with the purge supply and/or exhaust isolation valves open shall be limited to less than or equal to 250 hours during a calendar year and to less than or equal to 200 hours per calendar year while in MODES 1 or 2. The purge supply and exhaust isolation valves shall not be opened wider than 33 or 30 degrees, respectively (90 degrees is fully open).

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

ACTION:

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

b. With the containment purge supply and/or exhaust isolation valve(s) open for more than 250 hours during a calendar year, or for more than 200 hours during a calendar year while in MODES 1 or 2, close the valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION shall apply to both units simultaneously with the limits applicable to the sum of cumulative times for both units.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.8.3, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDBY within the next 6 hours, and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each containment purge supply and exhaust isolation valve shall be verified to be sealed closed at least once per 31 days.

4.6.1.7.2 The cumulative times that Unit 3 and 4 purge supply and/or exhaust valves have been open during the calendar year shall be determined at least once per 7 days.

4.6.1.7.3 At least once per refueling interval, not to exceed 24 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.20 L_a when pressurized to P_a.

4.6.1.7.4 For each containment purge supply or exhaust isolation valve that is not sealed closed, at least once per 24 hours verify that the valve is open no more than 33 or 30 degrees, respectively.

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CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the RWST.

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

ACTION:

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

b. With two Containment Spray Systems inoperable restore at least one Spray System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each Containment Spray System shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position and that power is available to flow path components that require power for operation;

b. By verifying that on recirculation flow, each pump develops the indicated differential pressure, when tested pursuant to Specification 4.0.5:

Containment Spray Pump ≥268 psid at a metered flowrate ≥400 gpm (Unit 3), or

≥269 psid while aligned in recirculation mode (Unit 4).

USE THIS INSERT FOR 3/4.6.1.7

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply (48 inch) and exhaust (54 inch) isolation valves shall be OPERABLE* and shall not be opened wider than 33 or 30 degrees, respectively (90 degrees is fully open).

APPLICABILITY: MODES 1, 2, 3, AND 4

ACTION:

1. With the purge supply and/or exhaust isolation valve(s) open wider than 33 or 30 degrees, respectively, within 4 hours close the valve(s) to less than or equal to 33 or 30 degrees, respectively~~,~~ or close the valve(s), or be in HOT STANDBY within next 6 hours and in COLD SHUTDOWN within the subsequent 30 hours.

2. With the purge supply and/or exhaust isolation valve(s) open in Modes 1 and 2 for more than 200 hours per site (Units 3 and 4) ~~unit~~ per calendar year, close the open valve(s) within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the subsequent 30 hours.

3. With the purge supply and/or exhaust isolation valve(s) having a measured leakage rate in excess of the limits of Specification 4.6.1.7.2 and the containment combined leakage for Type B and C penetrations is less than 0.60 L_a, restore the valve leakage within its limit during the next COLD SHUTDOWN condition of 72 hours or longer.

---

* ~~Each~~ The containment purge supply and exhaust isolation valve (s) total opening time in Modes 1 and 2 per site (Units 3 and 4) during a calendar year shall not exceed 200 hours unless a relief has been requested and received from the Commission.

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CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 The cumulative time that the Unit 3 and 4 purge supply and exhaust valves have been open in MODES 1 and 2 during the calendar year shall be determined at least once per 7 days.

4.6.1.7.2 At least once per refueling interval, not to exceed 24 months, each containment purge supply and exhaust isolation valve shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.20 L_a when pressurized to P_a.

4.6.1.7.3 At least once per refueling interval, not to exceed 24 months, verify that the mechanical stops for the containment purge supply and exhaust isolation valves are in place to ensure the valves will open no more than 33 or 30 degrees, respectively.

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SURVEILLANCE REQUIREMENTS (Continued)

not to exceed 24 months

c. At refueling by: #1

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

2) Verifying that each spray pump starts automatically on a containment spray actuation test signal. The manual isolation valves in the spray lines at the containment shall be locked closed for the performance of these tests.

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

PR

CONTAINMENT SYSTEMS

Emergency CONTAINMENT COOLING SYSTEM)E

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three emergency containment cooling units shall be OPERABLE.

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

ACTION:

a. With one of the above required emergency containment cooling units inoperable restore the inoperable cooling unit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.)E

b. With two or more of the above required emergency containment cooling units inoperable, restore at least two cooling ~~fans~~ units to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. |E

SURVEILLANCE REQUIREMENTS

4.6.2.~~3~~ 2 Each emergency containment cooling unit shall be demonstrated OPERABLE:)E

a. At least once per 31 days by: a

~~1)~~ a Starting each cooler ~~unit~~ fan from the control room and verifying that each fan ~~unit~~ motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes, ~~and~~ a)E

2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler. ≠1

b. At least once per refueling not to exceed 24 months by: 1) Verifying that each unit starts automatically on a safety injection (SI) test signal, and

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3 Three emergency containment filtering units shall be OPERABLE.

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

ACTION:

With one emergency containment filtering unit inoperable, restore the inoperable filter to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3 Each emergency containment filtering unit shall be demonstrated OPERABLE:

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

b. At least once per ~~18 months~~ refueling, not to exceed 24 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release or (3) after every 720 hours of system operation by: #1

   1) Performance of a visual inspection for foreign material and gasket deterioration, and verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP and halogenated hydrocarbons at the system flow rate of 37,500 cfm ±10%;

   2) Verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with applicable portions of Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and performed in accordance with ANSI N-510-1975, meets the acceptance criteria of greater than 99.9% removal of elemental iodine; and that any charcoal failing to meet this criteria be replaced with charcoal that meets or exceeds the criteria of position C.6a of Regulatory Guide 1.52, Rev. 2; and

   3) Verifying a system flow rate of 37,500 cfm ±10% and a pressure drop across the HEPA and charcoal filters of less than 6 inches water gauge during system operation when tested in accordance with ANSI N510-1975;

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SURVEILLANCE REQUIREMENTS (Continued)

c. After maintenance affecting flow distribution, by performance of a visual inspection and an air distribution test at a system flow rate of 37,500 cfm ±10%;

refueling, not to exceed 24 months

d. At least once per ~~18 months~~ by: #1

1) Verifying that the system starts on a Safety Injection test signal and;

2) Verifying that the filter cooling solenoid valves can be opened by operator action and are opened automatically on a loss of flow signal..

e. After each complete or partial replacement of a HEPA filter bank, by performance of a visual inspection for foreign material and gasket deterioration and by verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of DOP test aerosol while operating the system at a flow rate of 37,500 cfm ±10%; and E

f. After each complete or partial replacement of a charcoal adsorber bank, by performance of a visual inspection for foreign material and gasket deterioration and by verifying that the filtering unit satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% removal of halogenated hydrocarbon while operating the system at a flow rate of 37,500 cfm ±10%.

PR

CONTAINMENT SYSTEMS

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.~~) #2

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:

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

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

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

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. ~~(Specification 3.0.4 applies.)~~) E

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 which could affect cycling by performance of a cycling test, ~~and verification of isolation time.~~)#2)E

*CAUTION: The inoperable isolation valve(s) may be part of a system(s). Isolating the affected penetration(s) may affect the use of the system(s). Consider the technical specification requirements on the affected system(s) and act accordingly.

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SURVEILLANCE REQUIREMENTS (Continued) not to exceed 24 months

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 refueling by:) E

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

b. Verifying that on a Phase "B" Isolation test signal, each Phase "B" isolation valve actuates to its isolation position; and

c. Verifying that on a Containment Ventilation Isolation test signal, each purge, exhaust and instrument air bleed valve actuates to its isolation position. | E

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.~~) #2

No additional requirements other than those required by Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
A.	**Phase "A" Isolation**			
1.	CV-*-200A	Letdown Line	10	1,3
2.	CV-*-200B	Letdown Line	10	1,3
3.	CV-*-200C	Letdown Line	10	1,3
4.	CV-*-204	Letdown Line	10	1,3
5.	MOV-*-381	RCP Seal Water Leakoff Bypass	10	1,3
6.	CV-*-516	PRT Gas Analyzer Line	10	1,3
7.	CV-*-519A	PRT Makeup Primary Water Supply	60	1,3
9 ~~8~~.	CV-*-855	N_2 Supply to Accumulators	10	1,3
10 ~~9~~.	CV-*-956A	Pressurizer Steam Space Sample	10	1,3
11 ~~10~~.	CV-*-956B	Pressurizer Liquid Space Sample	10	1,3
12 ~~11~~.	CV-*-956D	Accumulator Sample Lines	10	1,3
13 ~~12~~.	MOV-*-1417	CCW to Normal CTMT Coolers	N/A	3,4,7
14 ~~13~~.	MOV-*-1418	CCW Return from Normal CTMT Coolers	N/A	3,4,7
~~14.~~	~~MOV-*-1425~~	~~S/G C Blowdown Sample~~	~~N/A~~	~~3,4,7~~
~~15.~~	~~MOV-*-1426~~	~~S/G B Blowdown Sample~~	~~N/A~~	~~3,4,7~~
~~16.~~	~~MOV-*-1427~~	~~S/G A Blowdown Sample~~	~~N/A~~	~~3,4,7~~

#1

#2

~~1 Subject to Type C Test of 10CFR50, Appendix J~~
~~2 Deleted~~
~~3~~ *-Applicable to Unit 3 and Unit 4
~~4 N/A - No specific closure time assumed in the LOCA accident analysis~~
~~5 May be opened on an intermittent basis under administrative control~~
~~6 Deleted~~
~~7 Isolation time requirement does not supercede requirement of IST program.~~

#3

8. CV-*-739 Excess Letdown Heat Ex Cooling Water Inlet

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

9 E

VALVE NUMBER		FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
A.		**Phase "A" Isolation (Continued)**		
15 ~~17~~.	CV-*-2821	Containment Sump Discharge	10	1,3
16 ~~18~~.	CV-*-2822	Containment Sump Discharge	10	1,3
17. ~~19~~.	SV-*-2911	Containment Air Sample	10	1,3
18. ~~20~~.	SV-*-2912	Containment Air Sample	10	1,3
19. ~~21~~.	SV-*-2913	Containment Air Sample	10	1,3
20. ~~22~~.	CV-*-4658A	RC Drain Tank Vent	10	1,3
21. ~~23~~.	CV-*-4658B	RC Drain Tank Vent	10	1,3
22. ~~24~~.	CV-*-4659A	RC Drain Tank Line to H_2 Analyzer	10	1,3
23. ~~25~~.	CV-*-4659B	RC Drain Tank Line to H^2 Analyzer	10	1,3
24. ~~26~~.	CV-*-4668A	RC Drain Tank Pump Discharge	10	1,3
25. ~~27~~.	CV-*-4668B	RC Drain Tank Pump Discharge	10	1,3
26. ~~28~~.	CV-*-6165	Breathing Air	10	1,3
27. ~~29~~.	SV-*-6385	PRT Gas Analyzer Line	10	1,3
28. ~~30~~.	MOV-*-6386	Excess Letdown and RCP Seal Water Return to CVCS	18	1,3
29. ~~31~~.	SV-*-6428	RCS Sample	10	1,3
32.	CV-*-6275A	S/G A Blowdown	N/A	3,4,7
33.	CV-*-6275B	S/G B Blowdown	N/A	3,4,7
34.	CV-*-6275C	S/G C Blowdown	N/A	3,4,7
35.	CV-*-6275A-1	S/G A Blowdown Bypass	N/A	3,4,7
36.	CV-*-6275B-1	S/G B Blowdown Bypass	N/A	3,4,7
37.	CV-*-6275C-1	S/G C Blowdown Bypass	N/A	3,4,7
~~38.~~	~~CV-*-739~~	~~Excess Letdown Heat Ex Cooling Water Outlet~~	~~N/A~~	~~3,4,7~~

#2

#1

~~1 Subject to Type C Test of 10CFR50, Appendix J~~
~~2 Deleted~~
~~3~~ *-Applicable to Unit 3 and Unit 4
~~4 N/A - No specific closure time assumed in the LOCA accident analysis~~
~~5 May be opened on an intermittent basis under administrative control~~
~~6 Deleted~~
~~7 Isolation time requirement does not supercede requirement of IST program.~~

#3

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TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SECONDS)	NOTES
B. Phase "B" Isolation			
1. ~~FCV~~ MOV-*-626	CC Return from RCP	N/A	3,4,7
2. MOV-*-716A	CC Supply to RCP	N/A	3,4,7
3. MOV-*-716B	CC Supply to RCP	N/A	3,4,7
4. MOV-*-730	CC Return from RCP Thermal Barrier Coolers	N/A	3,4,7
C. Containment Ventilation Isolation			
1. POV-*-2600	Containment Purge Supply	5	1,3
2. POV-*-2601	Containment Purge Supply	5	1,3
3. POV-*-2602	Containment Purge Exhaust	5	1,3
4. POV-*-2603	Containment Purge Exhaust	5	1,3
5. CV-*-2819	Instrument Air Bleed	10	1,3
6. CV-*-2826	Instrument Air Bleed	10	1,3

1E

#2

~~1 Subject to Type C Test of 10CFR50, Appendix J~~
~~2 Deleted~~
~~3~~ *-Applicable to Unit 3 and Unit 4
~~4 N/A - No specific closure time assumed in the LOCA accident analysis~~
~~5 May be opened on an intermittent basis under administrative control~~
~~6 Deleted~~
~~7 Isolation time requirement does not supercede requirement of IST program.~~

#3

P/

CONTAINMENT SYSTEMS

3/4.6.5 ~~COMBUSTIBLE GAS CONTROL~~

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.5 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

b. With both hydrogen monitors inoperable, restore at least one monitor to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

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

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

4.6.5.2 The flow path to each hydrogen monitor shall be demonstrated OPERABLE at least once per 31 days by a system walkdown to verify that each accessible manual, power operated, or automatic valve is in its correct position and that power is available to those components related to the operability of the flowpath.

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CONTAINMENT SYSTEMS

3/4.6.6 POST ACCIDENT CONTAINMENT VENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6 A Post Accident Containment Vent System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the Post Accident Containment Vent System inoperable, restore the Post Accident Containment Vent System to OPERABLE status within 7 days or be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.6 The Post Accident Containment Vent System shall be demonstrated OPERABLE:

a. At least once per 31 days by demonstrating system flow path operability via a system walkdown to verify that each accessible manual valve is in its correct position.

b. At least once per ~~18 months~~ refueling, not to exceed 24 months or (1) after any structural maintenance of the HEPA filter or charcoal adsorber housings, or (2) following operational exposure of filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system, or (3) after 720 hours of system operation or (4) after replacement of a filter by: #2

   1) A visual inspection of the system for foreign materials and gasket deterioration and verifying that the filter system satisfies the penetration and bypass leakage testing acceptance criteria of less than 1% for DOP and halogenated hydrocarbon tests conducted at a design flow rate of 55 cfm ±10%;

   2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample ~~obtained in accordance with Regulatory Position C.6.b of of Regulatory Guide 1.52, Revision 2, March 1978, and~~ performed in accordance with ANSI N510-1975, meets the methyl iodide removal criteria of greater than or equal to 90% and that any charcoal failing to meet the criteria be replaced with charcoal that meets or exceeds the criteria of Position C.6.a of Regulatory Guide 1.52, Revision 2. #1

PR

SURVEILLANCE REQUIREMENTS (Continued)

refueling not to exceed 24 months

c. At least once per ~~18 months~~ by: #2

1) Verifying that the pressure drop across the combined HEPA filter and charcoal adsorber is less than 6 inches Water Gauge at a flow rate of 55 cfm ± 10%,

2) Visual inspection of the system and operation of all valves.

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator ~~of an unisolated reactor coolant loop~~ shall be OPERABLE with lift settings as specified in Table 3.7-2. #1

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With (3) reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, operation in MODES 1, 2, and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.7-1

MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER RANGE NEUTRON FLUX HIGH SETPOINT (PERCENT OF RATED THERMAL POWER)
1	82
2	54
3	27

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER				LIFT SETTING (±1%)*	ORIFICE SIZE SQUARE INCHES
	Loop A	Loop B	Loop C		
1.	RV1400	RV1405	RV1410	1085 psig	16
2.	RV1401	RV1406	RV1411	1100 psig	16
3.	RV1402	RV1407	RV1412	1115 psig	16
4.	RV1403	RV1408	RV1413	1130 psig	16

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

rk

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent auxiliary feedwater trains including 3 pumps as specified in Table 3.7-3 and associated flowpaths shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

1) With one of the two required independent auxiliary feedwater trains inoperable, either restore the inoperable train to an OPERABLE status within 72 hours, or place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. #1

2) With both required auxiliary feedwater trains inoperable, within 2 hours either restore both trains to an OPERABLE status, or restore one train to an OPERABLE status and follow ACTION statement 1 above for the other train. If neither train can be restored to an OPERABLE status within 2 hours, verify the availability of both standby feedwater pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise, initiate corrective action to restore at least one auxiliary feedwater train to an OPERABLE status as soon as possible and follow ACTION statement 1 above for the other train. #1

3) With a single auxiliary feedwater pump inoperable, within 4 hours, verify OPERABILITY of two independent auxiliary feedwater trains, or follow ACTION statements 1 or 2 above as applicable. Upon verification of the OPERABILITY of two independent auxiliary feedwater trains, restore the inoperable auxiliary feedwater pump to an OPERABLE status within 30 days, or place the operating unit(s) in at least HOT STANDBY within 6 hours* and in HOT SHUTDOWN within the following 6 hours. The provisions of Specification 3.0.4 are not applicable during the 30 day period for the inoperable auxiliary feedwater pump. #1

SURVEILLANCE REQUIREMENTS

4.7.1.2.1 The required independent auxiliary feedwater trains shall be demonstrated OPERABLE:

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

1) Verifying by control panel indication and visual observation of equipment that each steam turbine-driven pump operates for 15 minutes or greater and develops a flow of greater than or

*If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. #

SURVEILLANCE REQUIREMENTS (Continued)

equal to 373 gpm to the entrance of the steam generators. The provisions of Specification 4.0.4 are not applicable for entry into MODES 2 and 3;

2) Verifying by control panel indication and visual observation of equipment that the auxiliary feedwater discharge valves and the steam supply and turbine pressure valves operate as required to deliver the required flow during the pump performance test above;

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

4) Verifying that power is available to those components which require power for ~~flow~~ path operability.

b. At least once per ~~18 months~~ refueling not to exceed 24 months by: #2

1) Verifying that each automatic valve in the flow path actuates to its correct position upon receipt of each Auxiliary Feedwater Actuation test signal, and

2) Verifying that each auxiliary feedwater pump receives a start signal as designed automatically upon receipt of each Auxiliary Feedwater Actuation test signal.

4.7.1.2.2 An auxiliary feedwater flow path to each steam generator shall be demonstrated OPERABLE following each COLD SHUTDOWN of greater than 30 days prior to entering MODE 1 by verifying normal flow to each steam generator.

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TABLE 3.7-3

AUXILIARY FEEDWATER SYSTEM OPERABILITY

UNIT	TRAIN	STEAM SUPPLY FLOWPATH(3)	PUMP	DISCHARGE WATER FLOWPATH(3)
3	1	SG 3C via MOV-3-1405(1) or SG 3B via MOV-3-1404(1)	A or C(2)	SG 3A via CV-3-2816 SG 3B via CV-3-2817 SG 3C via CV-3-2818
3	2	SG 3A via MOV-3-1403(1) or SG 3B via MOV-3-1404(1)	B or C(2)	SG 3A via CV-3-2831 SG 3B via CV-3-2832 SG 3C via CV-3-2833
4	1	SG 4C via MOV-4-1405(1) or SG 4B via MOV-4-1404(1)	A or C(2)	SG 4A via CV-4-2816 SG 4B via CV-4-2817 SG 4C via CV-4-2818
4	2	SG 4A via MOV-4-1403(1) or SG 4B via MOV-4-1404(1)	B or C(2).	SG 4A via CV-4-2831 SG 4B via CV-4-2832 SG 4C via CV-4-2833

NOTES:

(1) Steam admission valves MOV-3-1404 and MOV-4-1404 can be aligned to either train (but not both) to restore OPERABILITY in the event MOV-3-1403 or MOV-3-1405, or MOV-4-1403 or MOV-4-1405 are inoperable.

(2) During single and two unit operation, one pump shall be OPERABLE in each train and the third auxiliary feedwater pump shall be OPERABLE and capable of being ~~powered~~ from, and supplying water to either train, except as noted in ACTION ~~4~~ 3 of Technical Specification 3.7.1.2. The third auxiliary feedwater ~~pump~~ (normally the "C" pump) can be aligned to either train to restore OPERABILITY in the event one of the required pumps is inoperable. |E

(3) If any local manual realignment of valves is required when operating the auxiliary feedwater pumps, a dedicated individual, who is in communication with the control room, shall be stationed at the auxiliary pump area. Upon instructions from the control room, this operator would realign the valves in the AFW system train to its normal operational alignment. |E

feedwater

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PLANT SYSTEMS

3/4.7.1.3 CONDENSATE STORAGE TANK | E

LIMITING CONDITION FOR OPERATION

3.7.1.3 The Condensate Storage Tanks shall be OPERABLE with a contained water volume of at least 185,000 gallons of water as follows:

Single Unit Prior To Escalating into MODE 3
~~Opposite Unit in MODES 4, 5 or 6~~ | E

a) ONE water supply from either Condensate Storage Tank including flowpath piping and valves.

Second Unit Prior To Escalating into MODE 3
~~Opposite Unit in MODES 1, 2 or 3~~ | E

a) ONE water supply from ~~the~~ each unit's corresponding Condensate Storage Tank including flowpath piping and valves.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Single Unit At or Above MODE 3
~~Opposite Unit in MODES 4, 5 or 6~~ | E

1) With one water supply from a Condensate Storage Tank inoperable, within 4 hours, either realign the other Condensate Storage Tank containing the required water volume to the suction of the Auxiliary Feedwater pumps or restore the inoperable water supply to OPERABLE status or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

insert A

2) With both water supplies from the Condensate Storage Tanks inoperable, within 4 hours restore the water supply from either Condensate Storage Tank to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. | #1

Both Units At or Above MODE 3
~~Opposite Unit in MODES 1, 2 or 3~~ | E

1) With one water supply from a Condensate Storage Tank inoperable, restore the inoperable water supply to OPERABLE status within 4 hours or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. Refer to Single Unit Operation ACTION for single unit at or above MODE 3.

insert A

2) With both water supplies from the Condensate Storage Tank inoperable within 1 hour restore one water supply from a Condensate Storage Tank to OPERABLE status or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. If unable to restore at least one water supply from a Condensate Storage Tank to OPERABLE status within 4 hours from initial declaration of inoperability, the second unit shall be placed in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. | #1

(A) insert to 3.7.1.3

2) With both required Condensate Storage Tanks inoperable, within 2 hours either restore both tanks to an OPERABLE status, or restore one tank to an OPERABLE status and follow ACTION statement for the other tank. If neither tank can be restored to an OPERABLE status within 2 hours, verify the availability of both standby feedwater pumps and place the affected unit(s) in at least HOT STANDBY within the next 6 hours* and in HOT SHUTDOWN within the following 6 hours. Otherwise initiate corrective action to restore at least one Condensate Storage Tank to an OPERABLE status as soon as possible and follow ACTION statement above for the other tank.

* If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours #2

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SURVEILLANCE REQUIREMENTS (Continued)

4.7.1.3 The Condensate Storage Tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limit when the tank is the supply source for the auxiliary feedwater pumps.

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SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the Secondary Coolant System shall be less than or equal to 0.10 microCurie/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the Secondary Coolant System greater than 0.10 microCurie/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the Secondary Coolant System shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

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TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY

SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Radioactivity Determination	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) Once per 31 days, whenever the gross radioactivity determination indicates concentrations greater than 10% of the allowable limit for radioiodines. b) Once per 6 months, whenever the gross radioactivity determination indicates concentrations less than or equal to 10% of the allowable limit for radioiodines.

~~*A gross radioactivity analysis shall consist of the quantitative measurement of the total specific activity of the secondary coolant except for radionuclides with half-lives less than 10 minutes. Determination of the contributors to the gross specific activity shall be based upon those energy peaks identifiable with a 95% confidence level.~~ #1

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PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve (MSIV) shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

MODE 1:

With one MSIV inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 24 ~~4~~ hours; otherwise be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.)#1

at least

MODES 2 and 3:

With one MSIV inoperable, subsequent operation in MODE 2 or 3 may proceed provided the isolation valve is maintained closed. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.)E

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each MSIV shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.)#2

2 and

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PLANT SYSTEMS

STANDBY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two standby feedwater pumps shall be OPERABLE* and at least 60,000 gallons of water (available volume), shall be in the Demineralized Water Storage Tank**.

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one standby feedwater pump inoperable, restore the inoperable pumps to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6d.

b. With both standby feedwater pumps inoperable:

   1. Within 24 hours, notify the NRC and provide cause for inoperability ~~unavailability~~ and plans to restore pump(s) to ~~available~~ OPERABLE status and, E

   2. Submit a SPECIAL REPORT per 3.7.1.6d.

c. With less than 60,000 gallons of water in the Demineralized Water Storage Tank restore the available volume to at least 60,000 gallons within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the ~~unavailability,~~ inoperability action taken and a schedule for restoration within 30 days in accordance with 6.9.2. E

e. The provisions of Specification 3.0.3 are not applicable. #2

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 The Demineralized Water Storage tank water volume shall be determined to be within limits at least once per 24 hours.

4.7.1.6.2 At least monthly verify the standby feedwater pumps are OPERABLE by testing in recirculation on a STAGGERED TEST BASIS.

4.7.1.6.3 During each refueling not to exceed 24 months ~~outage,~~ verify operability of the respective standby feedwater pump by powering from the non-safety grade diesel generators and providing feedwater to the steam generators. #3

*These pumps are not safety related equipment and do not require plant safety related emergency power sources for operability, and the flowpath is normally isolated.
**The Demineralized Water Storage Tank is non-safety grade. #1

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PLANT SYSTEMS

3/4.7.2 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.2 The Component Cooling Water System (CCW) shall be OPERABLE with:

a. Three CCW pumps,

b. ~~Three~~ Two CCW heat exchangers in service* that are ~~such that each possible combination of two heat exchangers and one pump is~~ capable of removing design basis heat loads. ~~and~~ #1

~~c. All necessary valves, interlocks, and piping.~~ #2

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

ACTION: With the two inservice heat exchangers not capable of removing design bases heat loads, restore the capability within 1 hour

a. With only two CCW pumps with independent power supplies OPERABLE, restore the inoperable CCW pump to OPERABLE status within ~~7~~ 30 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable. #5

b. With only one CCW pump OPERABLE or with two CCW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within ~~24~~ 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. #3

c. ~~With one CCW heat exchanger inoperable, restore three heat exchangers to OPERABLE status within 72 hours~~ or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.** #1 #8

~~d. With any valves, interlocks or piping inoperable follow applicable action statements a., b., and c.~~ #2

SURVEILLANCE REQUIREMENTS

4.7.2 The Component Cooling Water System (CCW) shall be demonstrated OPERABLE: #4

a. At least once per 12 hours, by verifying that ~~each combination of two heat exchangers and~~ the two CCW heat exchangers in service, in conjunction with one pump, ~~is~~ are capable of removing design basis heat loads.

* Two heat exchangers, in conjunction with one pump, must be capable of removing design basis heat loads. #1

** If this ACTION applies to both units simultaneously, be in at least HOT STANDBY within the next 12 hours and in COLD SHUTDOWN within the following 30 hours. #8

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b. At least once per 31 days by: (1) verifying that each valve (manual, power-operated, or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position is in its correct position, and (2) verifying by a performance test the heat exchanger surveillance curves.

c. At least once per refueling, not to exceed 24 months ~~18 months during shutdown~~, by verifying that: #6.

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Component Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for CCW operability are OPERABLE.

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3/4.7.3 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 The Intake Cooling Water System (ICW) shall be OPERABLE with: #1

~~a.~~ three ICW pumps. ~~and two headers, and~~ #1+#2

~~b. All necessary valves, interlocks and piping.~~

APPLICABILITY: MODES 1, 2, 3, and 4. [ICW strainers – LATER] #6

ACTION:

a. With only two ICW pumps with independent power supplies OPERABLE, restore the inoperable ICW pump to OPERABLE status within 7 days or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable. #2

b. With only one ICW pump OPERABLE or with two ICW pumps OPERABLE but not from independent power supplies, restore two pumps from independent power supplies to OPERABLE status within ~~24~~ 72 hours or be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. E #3

SURVEILLANCE REQUIREMENTS

4.7.3 The Intake Cooling Water System (ICW) shall be demonstrated OPERABLE:

~~a. At least once per 12 hours by verifying that the system is capable of removing design basis heat loads for each single pump alignment.~~ #6

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

b ~~c~~. At least once per ~~18 months during shutdown~~ refueling, not to exceed 24 months, by verifying that: #4

1) Each automatic valve servicing safety-related equipment actuates to its correct position on a SI test signal, and

2) Each Intake Cooling Water System pump starts automatically on a SI test signal.

3) Interlocks required for system operability are OPERABLE.

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#1

3/4.7.4 ULTIMATE HEAT SINK

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LIMITING CONDITION FOR OPERATION

3.7.4 The ultimate heat sink shall be OPERABLE with an average supply water temperature to the Intake Cooling Water System less than or equal to 95°F.

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

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 12 hours and in COLD SHUTDOWN within the following 30 hours. This action shall be applicable to both units simultaneously.

SURVEILLANCE REQUIREMENTS

4.7.4 The ultimate heat sink shall be determined OPERABLE at least once per 24 hours by verifying the average supply water temperature* to the Intake Cooling Water System to be within its limit.

*Portable monitors may be used to measure the temperature.

PLANT SYSTEMS

3/4.7.5 CONTROL ROOM EMERGENCY VENTILATION SYSTEM E

LIMITING CONDITION FOR OPERATION

3.7.5 The Control Room Emergency Ventilation System shall be OPERABLE. E

APPLICABILITY: All MODES.

ACTION:

MODES 1, 2, 3 and 4:

With the Control Room Emergency Ventilation System inoperable, suspend all movement of fuel in the spent fuel pool and restore the inoperable system to OPERABLE status within 84 ~~72~~ hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. This ACTION shall apply to both units simultaneously.* E #3

MODES 5 and 6:

With the Control Room Emergency Ventilation System inoperable, suspend all operations involving CORE ALTERATIONS, movement of fuel in the spent fuel pool, or positive reactivity changes. This action shall apply to both units simultaneously. E

SURVEILLANCE REQUIREMENTS

4.7.5 The ~~Each~~ Control Room Emergency Ventilation System shall be demonstrated OPERABLE: #1

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F;

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

c. At least once per 18 months or (1) after 720 hours of system operation, or (2) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (3) following operational exposure of the filters to effluents from painting, fire, or chemical release in any ventilation zone communicating with the system, or (4) after complete or partial replacement of a filter bank by:

* If the ACTION applies to both units simultaneously, be in at [illegible] HOT STANDBY within the next [illegible] hours and in COLD SHUTDOWN within the following 30 hours. #4

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% DOP and halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%.)E

2) Verifying, within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ANSI N510-1975, meets the criteria for methyl iodine removal efficiency of greater than or equal to or 90% of the charcoal be replaced with charcoal that meets or exceeds the criteria of position C.6.a. of Regulatory Guide 1.52 (Revision 2), and /E

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration in the HEPA filters and charcoal adsorbers. #2

d. At least once per 12 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%;

e. At least once per 18 months by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation.

PLANT SYSTEMS

3/4.7.6 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.6 All snubbers shall be OPERABLE. The only snubbers excluded from the requirements are those installed on nonsafety-related systems and then only if their failure of failure of the system on which they are installed would have no adverse effect on any safety-related system.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable on any system, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.6g. on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system. 1E

SURVEILLANCE REQUIREMENTS 4.7.6f

4.7.6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these groups (inaccessible and accessible) may be inspected independently according to the schedule below. The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers. If all snubbers of each type (on any system) are found OPERABLE during the first inservice visual inspection, the second inservice visual inspection (of that system) shall be performed at the first refueling outage. Otherwise, subsequent visual inspections of a given system shall be performed in accordance with the following schedule: #2

No. of Inoperable Snubbers of Each Type (on any system) per Inspection Period	Subsequent Visual Inspection Period* **
0	~~18 months ± 25%~~ refueling, not to exceed 24 months
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%

#1

*The inspection interval for each type of snubber (on a given system) shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found (on that system).

**The provisions of Specification 4.0.2 are not applicable.

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SURVEILLANCE REQUIREMENTS (Continued)

#8

** Locked high radiation area (e.g., containment at power)

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that: (1) there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) fasteners for attachment of the snubber to the component and to the snubber anchorage are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that: (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per Specification 4.7.6f. ~~All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.~~ E | E

4.7.6e

d. Functional Tests

(not to exceed 24 months) #3

For each unit during refueling ~~shutdown~~, a representative sample of snubbers shall be tested using the following sample plan:

1) At least 10% of the total number of safety related snubbers for the respective unit identified by site records shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.6e, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested;

2) The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. At least 25% of the snubbers in the representative sample shall include snubbers from the following categories;

   A. Snubbers within 5 feet of heavy equipment (ex. valves, pumps, turbines, motors, etc.)

   B. Snubbers within 10 feet of the discharge from a safety relief valve.

3) Snubbers identified by site records as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.*

or

*Permanent ~~of~~ other exemptions from functional testing for individual snubbers in these ~~categories~~ may be granted by the Commission only if a justifiable basis ~~for exemption~~ is presented and/or snubber life destructive testing was performed to qualify snubber OPERABILITY for all design conditions at either the completion of their fabrication or at a subsequent date. | E

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1) Activation (restraining action) is achieved with the specified range of velocity or acceleration in both tension and compression;

2) Snubber release rate, where required, is within the specified range in tension and compression,

3) The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

f. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

If any snubber selected for functional testing either fails to activate or fails to move, i.e., frozen-in-place, the cause will be evaluated under the provisions of 10 CFR Part 21.

Should the results of the evaluation indicate that the failure was caused by either manufacturer or design deficiency, further action shall be taken, if needed, based on manufacturer or engineering recommendations.

For the snubber(s) found inoperable, an evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this evaluation shall be to determine if the components to which the inoperable snubber(s) are attached were adversely affected by the inoperability of the snubber(s) in order to ensure that the component remains capable of meeting the designed service.

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g. Snubber Service Life Monitoring Program

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.3m.

Concurrent with the first inservice visual inspection and during refueling shutdown thereafter, the installation and maintenance records for each safety related snubber as identified by site records shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This re-evaluation, replacement or reconditioning shall be indicated in the records.

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3/4.7.7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

   1. Decontaminate and repair the sealed source, or

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

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

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

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

   2) In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use; and

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

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

4.7.7.4 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

PLANT SYSTEMS

3/4.7.8 FIRE SUPPRESSION SYSTEMS

FIRE WATER SUPPLY AND DISTRIBUTION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 The Fire Water Supply and Distribution System shall be OPERABLE with:

a. At least two fire suppression pumps, one electric driven ~~with a rated capacity of 2000 gpm~~ and one diesel driven ~~with a rated capacity of 2500 gpm~~, with their discharge aligned to the fire suppression header, E

b. Two Separate water supplies, each with a minimum contained volume of 300,000 gallons, and E

c. An OPERABLE flow path capable of taking suction from the Raw Water Tank I and Raw Water Tank II and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe, and the last valve ahead of the deluge valve on each Deluge or Spray System required to be OPERABLE per Specifications 3.7.8.2, 3.7.8.~~4~~ 3, and 3.7.8.~~5~~ 4.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or provide an alternate backup pump or supply. The provisions of Specification 3.0.3 are not applicable. This action applies to both units simultaneously.

b. With the Fire Water Supply and Distribution System otherwise inoperable, establish a backup fire water capability within 24 hours. This action applies to both units simultaneously.

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PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.8.1.1 The Fire Water Supply and Distribution System shall be demonstrated OPERABLE:

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

b. At least once per 31 days by starting the electric motor-driven pump and operating it for at least 15 minutes on recirculation flow,

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

d. At least once per 12 months by performance of a system flush,

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

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

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

   2) Verifying that the electric-driven pump develops at least 1880 gpm at a system pressure of 130.2 psig, and the diesel-driven pump develops at least 2350 gpm at a system pressure of 130.2 psig E

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

   4) Verifying that each fire pump starts sequentially to maintain the Fire Water Supply and Distribution System pressure greater than or equal to 125 psig.

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

SURVEILLANCE REQUIREMENTS (Continued)

4.7.8.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

1) The fuel storage tank contains at least 375 gallons of fuel, and

2) The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-1975 is within the acceptable limits specified in Table 1 of ASTM D975-1977 when checked for viscosity and water and sediment; and

c. At least once per 18 months by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.8.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

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

1) The electrolyte level of each battery is above the plates, and

2) The overall battery voltage is greater than or equal to 24 volts.

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery, and

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

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

2) The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anticorrosion material.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.8.2 The following Spray and/or Sprinkler Systems shall be OPERABLE:

a. Fire Zones 47 and 54 - Component Cooling Water Areas ~~4160 V Switchgear Room louver spray~~

b. Fire Zones 45 and 55 - Charging Pump Rooms ~~Emergency Diesel Generator Building water current~~

c. Fire Zone 79A - North-South Breezeway ~~Control Point Guard House sprinkler system~~

d. Fire Zones 72, 73, 74 and 75 - Emergency Diesel Generator and Day Tank Rooms

#1

APPLICABILITY: Whenever equipment protected by the Spray/Sprinkler System is required to be OPERABLE.

ACTION:

a. With one or more of the above required Spray and/or Sprinkler Systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment. ~~Except for inoperability of the Switchgear Room louver spray~~ this action applies to both units simultaneously.)#1

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.2 Each of the above required Spray and/or Sprinkler Systems shall be demonstrated OPERABLE:

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

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

c. At least once per 18 months:

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

a) Verifying that the automatic valves in the flow path actuate to their correct positions on a ~~simulated fire~~ test signal, and)E

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

SURVEILLANCE REQUIREMENTS (Continued)

2) By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity; and

3) By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air or water ~~flow~~ test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed. E

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FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.8.3 The fire hose stations given in Table 3.7-4 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations given in Table 3.7-4 inoperable, provide an equivalent capacity fire hose from the nearest equivalent OPERABLE water source. The fire hose shall be of a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station, and shall be stored in a roll at the outlet of the OPERABLE water supply. The above ACTION requirement shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. This action applies to both units simultaneously. ~~for shared stations identified in Table 3.7.4.~~ |#1

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.3 Each of the fire hose stations given in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days, by a visual inspection of the fire hose stations accessible during plant operations to assure all required equipment is at the station.

b. At least once per 12 months, by:

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

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

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

c. At least once per 3 years, by:

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

   2) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

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TABLE 3.7-4

FIRE HOSE STATIONS

IDENTIFICATION	LOCATION	FIRE ZONE
HS-03-01*	EL. 18' - East of 4160V SWGR Room on Column	88
HS-03-02*	EL. 18' - West of 3A Condensate Pump on Pedestal	87
HS-03-03*	EL. 18' - Passageway South of SG Feed Pump Room	83
HS-03-04*	EL. 30' - East of 480V Load Center on Column	105
HS-03-05*	EL. 30' - South End of Mezzanine Deck	105
HS-03-06*	EL. 42' - NW End of Turbine Deck	117
HS-03-07*	EL. 42' - North of 6A HPFW Heater	117
HS-03-08*	EL. 42' - NW Corner of Entrance to Elevator	79
HS-04-01*	EL. 18' - South of 4160V SWGR Room on Column	82
HS-04-02*	EL. 18' - Passageway South of SG Feed Pump Room	78
HS-04-03*	EL. 30' - East of 480V Load Center at Stairway	105
HS-04-04*	EL. 30' - South End of Mezzanine Deck	105
HS-04-05*	EL. 42' - West End of Turbine Deck	117
HS-04-06*	EL. 42' - East Side of Turbine Deck and North of 6A FW Heater	117
HS-04-07*	EL. 42' - East Side of Turbine Deck and North of 6B FW Heater	117
HS-04-08*	EL. 42' - Southwest Corner of Turbine Deck	117
HS-AB-01*	EL. 18' - East-West Passageway at West End	58
HS-AB-02*	EL. 18' - East-West Passageway at East End	58
HS-AB-03*	EL. 18' - North-South Passageway Outside Unit 3 Charging Pump Room	58
HS-AB-04	EL. 50' - Roof of Unit 3 New Fuel Storage Area	118
HS-AB-05	EL. 50' - Roof of Unit 4 New Fuel Storage Area	118
HS-RW-01*	EL. 18' - Radwaste Bldg - Main Hallway	126
HS-RW-02*	EL. 38' - W of Control Room	126

#1

~~*Shared by Units 3 & 4.~~

#1

~~YARD~~ FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.8.4 The fire hydrants and associated hydrant hose houses given in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the ~~yard~~ fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the fire hydrants or associated hydrant hose houses given in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise, provide the additional hose within 24 hours.

b. The provisions of Specification 3.0.3 and are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.8.4 Each of the ~~yard~~ fire hydrants and associated hydrant hose houses given in Table 3.7-~~8~~ 5 shall be demonstrated OPERABLE:

a. At least once per 31 days, by visual inspection of the hydrant hose house to assure all required equipment is at the hose house,

b. At least once per 6 months by visually inspecting each fire hydrant and verifying that the hydrant is not damaged, and

c. At least once per 12 months by:

1) Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater,

2) Inspecting all the gaskets and replacing any degraded gaskets in the couplings, and

3) Performing a flow check of each hydrant to verify its OPERABILITY.

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TABLE 3.7-5

FIRE HYDRANTS

IDENTIFICATION	FIRE ZONE	LOCATION	NUMBER OF HYDRANTS
FH-01	124	NE Corner of Unit 3 near Vehicle Gate into RCA	1
FH-06	NA	W of Nuclear Maintenance Building	1
FH-07	86	Unit 3 Transformer Area	1
FH-08	81	Unit 4 Transformer Area	1
FH-09	76	Unit 4 Turbine-Generator Area	1
FH-10 FH-11	77	Unit 4 Condensate Storage Tank Area	2
FH-12	NA	Unit 4 New Fuel Storage Area	1
FH-13	123	Refueling Water Storage Area	1
FH-17	NA	Nuclear Dry Storage Area	1
FH-16	NA	Steam Generator Storage Area	1
		TOTAL	11

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3/4.7.9 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.9 All fire rated assemblies (walls, floor/ceilings, ~~cable~~ fire barrier penetration seals, ~~barriers,~~ and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire windows, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE. /E /E

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. The exposed surfaces of each fire rated assembly,

b. Each fire window/fire damper and associated hardware, and

c. At least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration will be inspected every 15 years.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.9.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the release and closing mechanism and latches at least once per 6 months, and by verifying:

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

b. That each locked closed fire door is closed at least once per 7 days,

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

d. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

Chapter 3/4.8 has been deleted from the Proof & Review version of the Revised Tech Specs. It will be submitted independently of the RTS project as a Proposed License Amendment to the current Tech Specs.

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of ~~0.90~~ 0.95 or less, or)#1

b. A boron concentration of greater than or equal to 1950 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 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or ~~its equivalent~~ until K_{eff} is reduced to less than or equal to ~~0.90~~ 0.95 or the boron concentration is restored to greater than or equal to 1950 ppm, whichever is the more restrictive. up to the flow capacity of one OPERABLE charging pump with suction from an OPERABLE RWST)#1 #2

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 vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 Valves isolating unborated water sources** shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

4.9.1.4 The spent fuel pit boron concentration shall be determined at least once per 31 days.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**The primary water supply to the boric acid blender may be opened under administrative controls for makeup.

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REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, one primary Source Range Neutron Flux Monitor with continuous visual indication in the control room and audible indication in the containment and control room, and one of the remaining three Source Range Neutron Flux Monitors (one primary or one of the two backup monitors) with continuous visual indication in the control room shall be OPERABLE.

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 required Source Range Neutron Flux Monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor vessel.

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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 ventilation 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 ventilation 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 ventilation isolation valves per the applicable portions of Specification 4.6.4.2.

*Exception may be taken under Administrative Controls for opening of certain valves and airlocks necessary to perform surveillance or testing requirements.

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 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1) A minimum capacity of 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 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 vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 At least once each refueling, each manipulator crane used for movement of fuel assemblies within the reactor 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 At least once each refueling, each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor 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.

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3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE AREAS

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.*

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, place the crane load in a safe condition.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Prior to crane operation over fuel assemblies in the spent fuel storage pool, verify that each load is 2000 pounds or less.

*Exception may be taken for the temporary construction crane to be used for the re-rack operation which may be carried over irradiated fuel to facilitate installation of the crane. Lift rigs which meet the design and operational requirements of NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" will be used while performing this installation.

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REFUELING OPERATIONS

3/4.9.8 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is greater than or equal to 23 feet.

ACTION:

With no RHR loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. #1

Immediately take action to

SURVEILLANCE REQUIREMENTS

4.9.8.1.1 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

4.9.8.1.2 The RHR flow indicator shall be subjected to a CHANNEL CALIBRATION at least once per refueling, not to exceed 24 months. #2

*The RHR 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 vessel hot legs.

REFUELING OPERATIONS

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 Two independent residual heat removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.

APPLICABILITY: MODE 6, when the water level above the top of the reactor vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor vessel flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Immediately take action to ~~C~~close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours. #1

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least one RHR loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

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3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The Containment Ventilation Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a. With the Containment Ventilation Isolation System inoperable, close each of the containment ventilation penetrations providing direct access from the containment atmosphere to the outside atmosphere.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.9 The Containment Ventilation 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 Ventilation Isolation occurs on a High Radiation test signal from ~~each of the~~ the required OPERABLE containment radiation monitoring instrumentation channel~~s~~. 1E #1

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3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the containment when either the fuel assemblies being moved or the fuel assemblies seated within the reactor vessel are irradiated while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

REFUELING OPERATIONS

3/4.9.11 WATER LEVEL - STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The water level shall be maintained ~~at elevation 57'0" (-2", +12")~~* greater than or equal to elevation 56'10" in the spent fuel storage pool.** #1

APPLICABILITY: Whenever irradiated fuel assemblies are in the storage pool.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

*During spent fuel rerack operation, the water level shall be maintained at least at 49'0" elevation. There will be no movement of fuel assemblies with water level lower than ~~57'0" (-2", +12")~~ 56'10" elevation during rerack operation. #1

**The requirements of this specification may be suspended for more than 4 hours hours to perform maintenance provided a safety evaluation is prepared prior to suspension of the above requirement and all movement of fuel assemblies and crane operation with loads in the fuel storage areas are suspended. If the level is not restored within 7 days, the NRC shall be notified within the next 24 hours.

FK

REFUELING OPERATIONS

3/4.9.12 HANDLING OF SPENT FUEL CASK

LIMITING CONDITION FOR OPERATION

3.9.12 The handling of spent fuel cask shall be limited to the following conditions:

1) The spent fuel cask shall not be moved into the spent fuel pit until all the spent fuel in the pit has decayed for a minimum of one thousand five hundred twenty-five (1,525) hours.*

2) Only a single element cask may be moved into the spent fuel pit.

3) A fuel assembly shall not be removed from the spent fuel pit in a shipping cask until it has decayed for a minimum of one hundred twenty (120) days.

APPLICABILITY: During movement of spent fuel cask in the spent fuel storage area.

ACTION:

With the requirement of the above specification not satisfied, suspend all movement of the spent fuel cask within the spent fuel storage area.

SURVEILLANCE REQUIREMENTS

4.9.12.1 The following required decay times of the spent fuel assemblies shall be determined prior to the movement of a spent fuel cask by verification of date and time the ~~spent fuel assemblies were placed into the spent fuel pit:~~ ~~the~~ most recently discharged fuel in the spent fuel pit was last critical: #

a. 1525 hours of decay of all spent fuel assemblies in the spent fuel pit for movement of a spent fuel cask into the spent fuel pit.

b. 120 days of decay of the spent fuel assembly which is in the spent fuel cask prior to removal of the spent fuel cask from the spent fuel pit.)#2

4.9.12.2 Prior to any operations involving spent fuel cask movement into the spent fuel pit, verify only a single element cask will be moved into the spent fuel pit.

4.8.12.3 The spent fuel cask crane interlock shall be demonstrated OPERABLE within 7 days of crane operation and at least once per 7 days (7 days is maximum time between tests; specification 4.0.2 does not apply here) when the crane is being used to maneuver the spent fuel cask.

---

*The spent fuel cask can be moved into the Unit 4 spent fuel pit after a minimum decay of 1000 hours until the new two-region high density spent fuel racks are installed.

FK

REFUELING OPERATIONS

3/4.9.13 RADIATION MONITORING

LIMITING CONDITION FOR OPERATION

3.9.13 ~~The~~ One Containment Radiation monitor~~s~~ which initiates containment and control room ventilation isolation shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

a) With ~~one or both~~ no radiation monitors ~~in~~operable, operation may continue provided the containment ventilation isolation valves are maintained closed, and

Use full caps #1

b) With ~~one or both~~ no radiation monitors ~~in~~operable, within 1 hour isolate the Control Room Emergency Ventilation System and initiate operation of the Control Room Emergency Ventilation System in the recirculation mode. E

SURVEILLANCE REQUIREMENTS

4.9.13 ~~Each~~ One Containment Radiation monitor required by Specification 3.9.13 shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies shown in Table 4.3-3. E

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. Fuel assemblies containing more than 4.1 weight percent of U-235 shall not be placed in the single region spent fuel storage racks. After installation of the two-region high density spent fuel racks, the maximum enrichment loading for the fuel assemblies in the spent fuel racks shall be 4.5 weight percent of U-235.

b. The minimum boron concentration in the Spent Fuel Pit shall be 1950 ppm.

c.* Storage in Region II of the Spent Fuel Pit shall be further restricted by burnup and enrichment limits specified in Table 3.9-1.

d.* During the re-racking operation only, fuel that does not meet the burnup requirement for normal storage in Region II may be stored in Region II in a checkerboard arrangement (i.e., no fuel stored in adjacent spaces).

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With any of conditions a, c or d not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

b. With boron concentration in the Spent Fuel Pit less than 1950 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 1950 ppm or greater.

SURVEILLANCE REQUIREMENTS

4.9.14 The boron concentration of the Spent Fuel Pit shall be verified to be 1950 ppm or greater at least once per month.

*These requirements are applicable only after installation of the new two-region high density spent fuel racks.

PR

TABLE 3.9-1

SPENT FUEL BURNUP REQUIREMENTS FOR STORAGE IN REGION II OF THE SPENT FUEL PIT

Initial w/o	Discharge Burnup GWD/MT
1.5	0.
1.75	5.0
2.0	9.0
2.2	12.0
2.4	14.8
2.6	17.6
2.8	20.1
3.0	22.6
3.2	25.0
3.4	27.4
3.6	29.6
3.8	31.8
4.0	34.0
4.2	36.1
4.5	39.0

Linear interpolation between two consecutive points will yield conservative results.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s).

APPLICABILITY: MODE 2.

ACTION:

a. With any full-length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at greater than or equal to 10 gpm of a solution containing greater than or equal to 20,000 ppm boron ~~or its equivalent~~ until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored. #1

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 10 gpm of a solution containing greater than or equal to 20,000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

up to the flow capacity of one OPERABLE charging charging pump with with suction from an OPERABLE RWST

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full-length control rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full-length control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

PK

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion, and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specification 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The requirements of the below listed specifications shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Specifications 4.2.2.1 and ~~4.2.2.5, and~~ E

b. Specification 4.2.3.~~2~~.3

SPECIAL TEST EXCEPTIONS

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5, and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range channels are set at less than or equal to 25% of RATED THERMAL POWER, and

c. The Reactor Coolant System lowest average operating loop temperature (T_{avg}) is greater than or equal to 531°F.)E

APPLICABILITY: MODE 2.

ACTION:

reduce RATED THERMAL POWER below 5% or

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the Reactor trip breakers.)#1

b. With a Reactor Coolant System average operating loop temperature ~~(T_{avg})~~ less than 531°F, restore ~~T_{avg}~~ RCS average temperature to within its limit within 15 minutes or be in at least HOT STANDBY within the next 15 minutes.)E

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System average temperature ~~(T_{avg})~~ shall be determined to be greater than or equal to 531°F at least once per 30 minutes during PHYSICS TESTS.)E

~~3/4.10.4 (This specification number is not used)~~

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.~~5~~ 4 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; | E

a. Only ~~one~~ two shutdown or control banks ~~is~~ are withdrawn from the fully inserted position at a time, and |#1|

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 Systems inoperable or with more than ~~one~~ Two banks of rods withdrawn, immediately open the Reactor trip breakers. |#1

SURVEILLANCE REQUIREMENTS

4.10.5 The above required 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 Digital Rod Position Indication System agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

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PR

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 in liquid effluents to UNRESTRICTED AREAS (see Figure 5.1-1) 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} microCurie/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

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 the radioactivity analyses shall be used in accordance with the methodology and parameters in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

PR

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)(1) (µCi/ml)
1. Batch Waste Release Tanks(2)	P Each Batch	P Each Batch	Principal Gamma Emitters(3)	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(4)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite(4)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
2. Continuous Releases(5)	W	W	Principal Gamma Emitters(3)	5×10^{-7}
			I-131	1×10^{-6}
a. Steam Generator Blowdown(7)	M(8)	M(8)	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	W(8)	M(8) Composite(6)	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	W(8)	Q(8) Composite(6)	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
b. Storm ~~Steame~~ Drain	M	M	Principal Gamma Emitters (3)	5×10^{-7}
			I-131	1×10^{-6}

E

TABLE NOTATIONS

(1)The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot (2.22 \times 10^6) \cdot Y \cdot [\exp(-\lambda \Delta t)]}$$

Where:

LLD = the "a priori" lower limit of detection as defined above for a blank sample (microCurie per unit mass or volume),

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22×10^6 = the number of disintegrations per minute per microCurie,

Y = the fractional radiochemical yield, when applicable,

λ = the radioactive decay constant for the particular radionuclide, and

Δt = the elapsed time between the midpoint of sample collection and the 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 should be used in the calculation.

(2)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.

TABLE NOTATIONS (Continued)

(3) The principal gamma emitters for which the LLD specification exclusively applies 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 considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Semiannual Radioactive Effluent Release Report pursuant to Specification 6.9.1.4.

(4) 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 that is representative of the liquids released.

(5) A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

(6) Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

(7) Sampling and analysis of steam generator blowdown is not required during Mode 5 or 6.

(8) Sampling and analysis of steam generator blowdown on the applicable unit is only necessary for these species when primary to-secondary leakage is occurring as indicated by the condenser air ejector noble gas activity monitor. (See Specification 3.3.3.7 in Table 3.3-8, item 3a).

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the whole body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the whole body and to less than or equal to 10 mrems 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, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.2 Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

LIQUID RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.1.3 The Liquid Radwaste Treatment System shall be OPERABLE and appropriate portions of the system shall be used to reduce releases of radioactivity when the projected doses due to the liquid effluent, from each unit, to UNRESTRICTED AREAS (see Figure 5.1-1) would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period.

APPLICABILITY: At all times.

ACTION:

a. With radioactive liquid waste being discharged without treatment and in excess of the above limits and any portion of the Liquid Radwaste Treatment System not in operation, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Explanation of why liquid radwaste was being discharged without treatment, identification of any inoperable equipment or subsystems, and the reason for the 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 Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases from each unit to UNRESTRICTED AREAS shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Liquid Radwaste Treatment Systems are not being fully utilized.

4.11.1.3.2 The installed Liquid Radwaste Treatment System shall be considered OPERABLE by meeting Specifications 3.11.1.1 and 3.11.1.2.

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 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the whole body and less than or equal to 3000 mrems/yr to the skin, and

b. For Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately restore 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 methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in 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 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)(1) (μCi/cc)
1. Gas Decay Tank (Batch)	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters(2)	1×10^{-4}
2. Containment Purge ~~or Venting~~ (Batch)	P(6) Grab Sample	P Each PURGE(6)	Principal Gamma Emitters(2)	1×10^{-4}
			H-3	1×10^{-6}
3. Condenser Air Ejectors	M(6) Grab Sample	M(6) Gas Sample	Principal Gamma Emitters(2)	1×10^{-4}
			H-3	1×10^{-6}
4. ~~Plant Vent (The plant vent incorporates Unit 4 Spent Fuel Pit Building Vent.)~~	M(6) Grab Sample	M(6) Gas Sample	Principal Gamma Emitters(2)	1×10^{-4}
	M(4),(5) Grab Sample	M	H-3	1×10^{-6}
5. Unit 3 Spent Fuel Pit Building Vent	M(5) Grab Sample	M Gas Sample	Principal Gamma Emitters(2)	1×10^{-4}
	M(4),(5) Grab Sample	M	H-3	1×10^{-6}
6. All Release Types as listed in 3., 4., and 5. above	Continuous(3)	W(7) Charcoal Sample	I-131	1×10^{-12}
	Continuous(3)	W(7) Particulate Sample	Principal Gamma Emitters(2)	1×10^{-11}
	Continuous(3)	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous(3)	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Cont[illegible]uous(3)	Noble Gas Monitor	[illegible]ble Gas [illegible]s Beta or Gamma	1×10^{-6}

4. Plant Vent incorporates
a. Unit 4 SFP Building
b. Containment(s) Venting
(Instrument Air Bleed Valves)

#1

TABLE 4.11-2 (Continued)

TABLE NOTATIONS

(1)The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot (2.22 \times 10^6) \cdot Y \cdot [\exp(-\lambda \Delta t)]}$$

Where:

LLD	=	the "a priori" lower limit of detection as defined above as a blank sample (microCurie per unit mass or volume),
s_b	=	the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),
E	=	the counting efficiency (counts per disintegration)
V	=	the sample size (units of mass or volume),
2.22×10^6	=	the number of disintegrations per minute per microCurie,
Y	=	the fractional radiochemical yield, when applicable,
λ	=	the radioactive decay constant for the particular radionuclide, and
Δt	=	the elapsed time between the midpoint of sample collection and the 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.

PK

TABLE 4.11-2 (Continued)

TABLE NOTATIONS (Continued)

(2)The principal gamma emitters for which the LLD specification will apply are exclusively the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 in noble gas emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141 and Ce-144 in for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other gamma peaks that are measurable and identifiable, together with the above nuclides, shall also be identified and reported pursuant to Specification 6.9.1.4.

Nuclides which are below the LLD for the analyses should not be reported as being present at the LLD for that nuclide. When a radionuclide's calculated LLD is greater than its listed LLD limit, the calculated LLD should be assigned as the activity of the radionuclide; or, the activity of the radionuclide should be calculated using measured ratios with those radionuclides which are routinely identified and measured.

(3)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.

(4)When a Unit's refueling canal is flooded Tritium grab samples shall be taken on that Unit only from the following respective area(s) at least once per 24 hours:

For Unit 3 sample the plant vent and the Unit 3 spent fuel pool area ventilation exhaust.

For Unit 4 sample the plant vent only.

(5)When spent fuel is in a Unit's spent fuel pool, tritium grab samples shall be taken on that unit only from the following respective area at least once per 7 days: | E

For Unit 3, ~~sample~~ the Unit 3 spent ~~fuel~~ pool area ventilation exhaust

For Unit 4, sample the plant vent.only. | E

(6)Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased by more than a factor of 3; ~~or~~ and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor of 3.) #2

TABLE NOTATIONS (Continued)

(7) Sample collection media on the applicable Unit shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing, or after removal from sampler. Sample collection media on the applicable Unit shall also be changed at least once per 24 hours for at least 7 days following each shutdown, startup, or THERMAL POWER change exceeding 15% of RATED THERMAL POWER within a 1-hour period and analyses shall be completed within 48 hours of changing & ~~This requirement does not apply~~ if: (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has ~~not~~ increased more than a factor of 3; and (2) the noble gas monitor shows that effluent activity has ~~not~~ increased more than a factor of 3. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. #3

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 unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation, and

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.2 Cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of Iodine-131, Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for Iodine-131, Iodine-133, tritium and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

PR

GASEOUS RADWASTE TREATMENT SYSTEM

LIMITING CONDITION FOR OPERATION

3.11.2.4 The VENTILATION EXHAUST TREATMENT SYSTEM and the GAS DECAY TANK SYSTEM shall be OPERABLE and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) would exceed:

a. 0.2 mrad to air from gamma radiation, or

b. 0.4 mrad to air from beta radiation, or

c. 0.3 mrem to any organ of a MEMBER OF THE PUBLIC.

APPLICABILITY: At all times.

ACTION:

a. With radioactive gaseous waste being discharged without treatment and in excess of the above limits, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that includes the following information:

   1. Identification of any inoperable equipment or subsystems, and the reason for the 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 Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from each unit to areas at and beyond the SITE BOUNDARY shall be projected at least once per 31 days in accordance with the methodology and parameters in the ODCM when Gaseous Radwaste Treatment Systems are not being fully utilized.

4.11.2.4.2 The installed VENTILATION EXHAUST TREATMENT SYSTEM and GAS DECAY TANK SYSTEM shall be considered OPERABLE by meeting Specifications 3.11.2.1 and 3.11.2.2 or 3.11.2.3.

F12

RADIOACTIVE EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the GAS DECAY TANK SYSTEM (as measured in the inservice Gas Decay Tank) shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.)E

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the inservice GAS DECAY TANK greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.)E

b. With the concentration of oxygen in the inservice GAS DECAY TANK greater than 4% by volume and the hydrogen concentration greater than 4% by volume, immediately suspend all additions of waste gases to the Gas Decay Tanks and reduce the concentration of oxygen to less than or equal to 4% by volume, then take ACTION a., above.)E)E

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and oxygen in the inservice GAS DECAY TANKS shall be determined to be within the above limits by continuously* monitoring the waste gases in the in-service ~~GAS DECAY TANK SYSTEM~~ with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-8 of Specification 3.3.3.7.)E)#2)#1

* When continuous monitoring capability is inoperable, Table 3.3-8 allows the use of grab samples.)#2

RADIOACTIVE EFFLUENTS

GAS DECAY TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas decay tank shall be limited to less than or equal to 70,000 Curies of noble gases (considered as Xe-133 equivalent).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas decay tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank, within 48 hours reduce the tank contents to within the limit, and describe the events leading to this condition in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.4.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas decay tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank and the Reactor Coolant System total activity exceeds the limit of Specification 3.4.8.

3/4.11.3 SOLID RADIOACTIVE WASTES

LIMITING CONDITION FOR OPERATION

SOLIDIFICATION or dewatering

if not

3.11.3 Radioactive wastes shall be ~~solidified or dewatered~~ performed in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and ~~disposal site~~ the applicable licensing requirements of the consignee when received at the ~~disposal site.~~ shipping destination.

#1

APPLICABILITY: At all times.

ACTION:

a. With SOLIDIFICATION or ~~dewatering~~ not meeting ~~disposal site and~~ the applicable licensing requirements of the consignee shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the PROCESS CONTROL PROGRAM, the procedures, and/or the Solid Waste System as necessary to prevent recurrence.

b. With SOLIDIFICATION or dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, test the improperly processed waste in each container to ensure that it meets ~~burial ground~~ the applicable licensing requirements of the consignee and shipping requirements and take appropriate ~~administrative action to prevent~~ recurrence.

c. ~~The provisions of Specification 3.0.3 are not applicable.~~

SURVEILLANCE REQUIREMENTS

4.11.3.1 ~~SOLIDIFICATION (dewatering only)~~ Dewatering shall be performed in accordance with the PCP.

4.11.3.2 SOLIDIFICATION (excluding dewatering) of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions) shall be verified in accordance with the PROCESS CONTROL PROGRAM:

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 three 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; and

c. With the installed equipment incapable of meeting Specification 3.11.3 or declared inoperable, restore the equipment to OPERABLE status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

RADIOACTIVE EFFLUENTS

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

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 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the units to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.11.4.1 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 methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the units and the methodology used shall be indicated in the Semiannual Radioactive Effluent Release Report. This requirement is applicable only under conditions set forth in ACTION a. of Specification 3.11.4.

PR

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The Radiological Environmental Monitoring Program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the Radiological Environmental Monitoring Program not being conducted as specified in Table 3.12-1, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of confirmed** radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose* to a MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, or 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

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 a MEMBER OF THE PUBLIC from all radionuclides is equal to or greater than the calendar year limits of Specification 3.11.1.2, 3.11.2.2, or 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 required by Specification 6.9.1.3.

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*The methodology and parameters used to estimate the potential annual dose to a MEMBER OF THE PUBLIC shall be indicated in this report.

**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 30 days.

PR

RADIOLOGICAL ENVIRONMENTAL MONITORING

LIMITING CONDITION FOR OPERATION

ACTION (Continued)

c. With milk or fresh leafy vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify specific locations for obtaining replacement samples and add them within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for the ODCM reflecting the new location(s) with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM[1]

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[2][3]	SAMPLING AND COLLECTION FREQUENCY[4]	TYPE AND FREQUENCY OF ANALYSIS[4]
1. Direct Radiation[5]	21 monitoring locations	Continuous monitoring with sample collection quarterly [6]	Gamma exposure rate quarterly
2. Airborne			
Radioiodine and Particulates	Five locations	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	Radioiodine Filter I-131 analysis weekly. Particulate Filter Gross beta radioactivity analysis $\geq$ 24 hours following filter change;[7] Gamma isotopic analysis[8] of composite[7] (by location) quarterly.
3. Waterborne[10]			
a. Surface[8]	Three locations[9]	Monthly	Gamma isotopic[8] and tritium analyses monthly.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM[1]

EXPOSURE PATHWAY AND/OR SAMPLE	NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS[2][3]	SAMPLING AND COLLECTION FREQUENCY[4]	TYPE AND FREQUENCY OF ANALYSIS[4]
3. Waterborne (Continued)			
b. Sediment from Shoreline	Three locations	Semiannually.	Gamma isotopic analysis[8] semiannually.
4. Ingestion			
a. Fish and Invertebrates			
1. Crustacea	Two locations	Semiannually	Gamma isotopic analysis[8] semiannually
2. Fish	Two locations	Semiannually	Gamma isotopic analysis [8] semiannually
b. Food Products			
1. Fresh leafy ~~Broad leaf~~ vegetation	Three locations[11]	Monthly when available	Gamma isotopic [8] and I-131 analyses monthly.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS

(1) Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of automatic sampling equipment or other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction, corrective action shall be taken prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(2) Specific parameters of distance and direction sector from the centerline of the plant vent stack and additional description where pertinent, shall be provided for each and every sample location in Table 3.12-1 in a table and figure(s) in the ODCM.

(3) At times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM.

(4) The following definition of frequencies shall apply to Table 3.12-1 only:

Weekly - not less than once per calendar week. A maximum interval of 11 days is allowed between the collection of any two consecutive samples.

Semi-Monthly - Not less than 2 times per calendar month with an interval of not less than 7 days between sample collections. A maximum interval of 24 days is allowed between collection of any two consecutive samples.

Monthly - Not less than once per calendar month with an interval of not less than 10 days between collection of any two consecutive samples.

Quarterly - Not less than once per calendar quarter.

Semiannually - One sample each between calendar dates (January 1 - June 30) and (July 1 - December 31). An interval of not less than 30 days will be provided between sample collections.

The frequency of analyses is to be consistent with the sample collection frequency.

(5) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For the purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters.

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TABLE 3.12-1 (Continued)

TABLE NOTATIONS (Continued)

(6) Refers to normal collection frequency. More frequent sample collection is permitted when conditions warrant it.

(7) Airborne particulate sample filters are analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. ~~If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.~~ In addition to the requirement for a gamma isotopic on a composite sample, a gamma isotopic is also required for each sample having a gross beta radioactivity which is > 1.0 pCi/m^3 and which is also > 10 times that of the most recent control sample. #1

(8) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

(9) Off-shore grab samples.

(10) Discharges from the Turkey Point Plant do not influence drinking water or ground water pathways.

(11) Samples of fresh leafy ~~broad leaf~~ vegetation grown nearest each of two different off-site locations of highest predicted annual average ground level D/Q, and one sample of similar ~~broad leaf~~ fresh leafy vegetation at an available location 15-30 km distant in the least prevalent wind direction based upon historical data in the ODCM. E

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TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m^3)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)
H-3	30,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95***	400				
I-131	2**	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140***	200			300	

*Since no drinking water pathway exists, a value of 30,000 pCi/l is used. For drinking water samples, a value of 20,000 pCi/l is used. This is 40 CFR Part 141 value.

**Applies to drinking water

***An equilibrium mixture of the parent and daughter isotopes which corresponds to the reporting value of the parent isotope.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS(1)

LOWER LIMIT OF DETECTION (LLD)(2)(3)

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m³)	FISH (pCi/kg, wet)	MILK (pCi/l)	FOOD PRODUCTS (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	3000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15(5)					
I-131	1(4)	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15(5)			15(5)		

*Since no drinking water pathway exists, a value of 3,000 pCi/l is used. For drinking water samples, a value of 2,000 pCi/l is used.

TABLE NOTATIONS

(1)This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(2)Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

(3)The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ (s_b)}{E \cdot V \cdot 2.22 \cdot Y \cdot [\exp(-\lambda\Delta t)]}$$

Where:

LLD = the "a priori" lower limit of detection as defined above as picoCuries per unit mass or volume,

s_b = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute),

E = the counting efficiency (counts per disintegration),

V = the sample size (units of mass or volume),

2.22 = the number of disintegrations per minute per picoCurie,

Y = the fractional radiochemical ~~yield, when~~ applicable,

λ = the radioactive decay constant for the particular radionuclide, and

Δt = for environmental samples is ~~if~~ the elapsed time between same collection, or end of the sample collection period, and time of counting ~~(for plant effluents, not ... 0)~~ E

Typical values ~~of~~ E, V, ~~Y,~~ and Δt ~~should be~~ used in the calculation.

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TABLE 4.12-1 (Continued)

TABLE NOTATIONS (Continued)

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. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

(4)LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

(5)An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/l of the parent isotope.

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RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A Land Use Census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, ~~and~~ the nearest garden* of greater than 50 m^2 (500 ft^2) producing ~~broad leaf~~ fresh leafy vegetation. E

APPLICABILITY: At all times.

ACTION:

a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, pursuant to Specification 6.9.1.4, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

b. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in the ODCM. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), 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. Pursuant to Specification 6.14, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table(s) for the ODCM reflecting the new location(s) with information supporting the change in sampling locations.

c. The provisions of Specification 3.0.3 are not applicable.

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*~~Broad leaf~~ Fresh leafy vegetation sampling may be performed at the SITE BOUNDARY in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for ~~broad leaf~~ fresh leafy vegetation sampling in Table 3.12-1, Part 4.b., shall be followed, including analysis of control samples. E

RADIOLOGICAL ENVIRONMENTAL MONITORING

SURVEILLANCE REQUIREMENTS

4.12.2 The Land Use Census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the Land Use Census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on all radioactive materials*, supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission**. #1

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 pursuant to Specification 6.9.1.3.

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.3.

* This specification shall apply only to those materials that represent the sampled media and for those analyses and isotopes listed in Tables 3.12-1 and 3.12-2. #1

**This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

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BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

<u>NOTE</u>

The BASES contained in succeeding pages summarize the reasons for the Specifications in Sections 3.0 and 4.0, but in accordance with 10 CFR 50.36 are not part of these Technical Specifications.

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BASES

Specification 3.0.1 through 3.0.4 establish the general requirements applicable to Limiting Conditions for Operation. These requirements are based on the requirements for Limiting Conditions for Operation stated in the Code of Federal Regulations, 10 CFR 50.36(c)(2):

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specification until the condition can be met."

Specification 3.0.1 establishes the Applicability statement within each individual specification as the requirement for when (i.e., in which OPERATIONAL MODES or other specified conditions) conformance to the Limiting Conditions for Operation is required for safe operation of the facility. The ACTION requirements establish those remedial measures that must be taken within specified time limits when the requirements of a Limiting Condition for Operation are not met.

There are two basic types of ACTION requirements. The first specifies the remedial measures that permit continued operation of the facility which is not further restricted by the time limits of the ACTION requirements. In this case, conformance to the ACTION requirements provides an acceptable level of safety for unlimited continued operation as long as the ACTION requirements continue to be met. The second type of ACTION requirement specifies a time limit in which conformance to the conditions of the Limiting Condition for Operation must be met. This time limit is the allowable outage time to restore an inoperable system or component to OPERABLE status or for restoring parameters within specified limits. If these actions are not completed within the allowable outage time limits, a shutdown is required to place the facility in a MODE or condition in which the specification no longer applies. It is not intended that the shutdown ACTION requirements be used as an operational convenience which permits (routine) voluntary removal of a system(s) or component(s) from service in lieu of other alternatives that would not result in redundant systems or components being inoperable.

The specified time limits of the ACTION requirements are applicable from the point in time it is identified that a Limiting Condition for Operation is not met. The time limits of the ACTION requirements are also applicable when a system or component is removed from service for surveillance testing or investigation of operational problems. Individual specifications may include a specified time limit for the completion of a Surveillance Requirement when equipment is removed from service. In this case, the allowable outage time limits of the ACTION requirements are applicable when this limit expires if the surveillance has not been completed. When a shutdown is required to comply with ACTION requirements, the plant may have entered a MODE in which a new specification becomes applicable. In this case, the time limits of the ACTION

requirements would apply from the point in time that the new specification becomes applicable if the requirements of the Limiting Condition for Operation are not met.

Specification 3.0.2 establishes that noncompliance with a specification exists when the requirements of the Limiting Condition for Operation are not met and the associated ACTION requirements have not been implemented within the specified time interval. The purpose of this specification is to clarify that (1) implementation of the ACTION requirements within the specified time interval constitutes compliance with a specification and (2) completion of the remedial measures of the ACTION requirements is not required when compliance with a Limiting Condition of Operation is restored within the time interval specified in the associated ACTION requirements.

Specification 3.0.3 establishes the shutdown ACTION requirements that must be implemented when a Limiting Condition for Operation is not met and the condition is not specifically addressed by the associated ACTION requirements. The purpose of this specification is to delineate the time limits for placing the unit in a safe shutdown MODE when plant operation cannot be maintained within the limits for safe operation defined by the Limiting Conditions for Operation and its ACTION requirements. It is not intended to be used as an operational convenience which permits (routine) voluntary removal of redundant systems or components from service in lieu of other alternatives that would not result in redundant systems or components being inoperable. One hour is allowed to prepare for an orderly shutdown before initiating a change in plant operation. This time permits the operator to coordinate the reduction in electrical generation with the load dispatcher to ensure the stability and availability of the electrical grid. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the cooldown capabilities of the facility assuming only the minimum required equipment is OPERABLE. This reduces thermal stresses on components of the primary coolant system and the potential for a plant upset that could challenge safety systems under conditions for which this specification applies.

If remedial measures permitting limited continued operation of the facility under the provisions of the ACTION requirements are completed, the shutdown may be terminated. The time limits of the ACTION requirements are applicable from the point in time there was a failure to meet a Limiting Condition for Operation. Therefore, the shutdown may be terminated if the ACTION requirements have been met or the time limits of the ACTION requirements have not expired, thus providing an allowance for the completion of the required actions.

The time limits of Specification 3.0.3 allow 37 hours for the plant to be in the COLD SHUTDOWN MODE when a shutdown is required during the POWER MODE of operation. If the plant is in a lower MODE of operation when a shutdown is required, the time limit for reaching the next lower MODE of operation

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applies. However, if a lower MODE of operation is reached in less time than allowed, the total allowable time to reach COLD SHUTDOWN, or other applicable MODE, is not reduced. For example, if HOT STANDBY is reached in 2 hours, the time allowed to reach HOT SHUTDOWN is the next 11 hours because the total time to reach HOT SHUTDOWN is not reduced from the allowable limit of 13 hours. Therefore, if remedial measures are completed that would permit a return to POWER operation, a penalty is not incurred by having to reach a lower MODE of operation in less than the total time allowed.

The same principle applies with regard to the allowable outage time limits of the ACTION requirements, if compliance with the ACTION requirements for one specification results in entry into a MODE or condition of operation for another specification in which the requirements of the Limiting Condition for Operation are not met. If the new specification becomes applicable in less time than specified, the difference may be added to the allowable outage time limits of the second specification. However, the allowable outage time limits of ACTION requirements for a higher MODE of operation may not be used to extend the allowable outage time that is applicable when a Limiting Condition for Operation is not met in a lower MODE of operation.

The shutdown requirements of Specification 3.0.3 do not apply in MODES 5 and 6, because the ACTION requirements of individual specifications define the remedial measures to be taken.

Specification 3.0.4 establishes limitations on MODE changes when a Limiting Condition for Operation is not met. It precludes placing the facility in a higher MODE of operation when the requirements for a Limiting Condition for Operation are not met and continued noncompliance to these conditions would result in a shutdown to comply with the ACTION requirements if a change in MODES were permitted. The purpose of this specification is to ensure that facility operation is not initiated or that higher MODES of operation ~~are~~ not entered when corrective action is being taken to obtain compliance with a specification by restoring equipment to OPERABLE status or parameters to specified limits. Compliance with ACTION requirements that permit continued operation of the facility for an unlimited period of time provides an acceptable level of safety for continued operation without regard to the status of the plant before or after a MODE change. Therefore, in this case, entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the provisions of the ACTION requirements. The provisions of this specification should not, however, be interpreted as endorsing the failure to exercise good practice in restoring systems or components to OPERABLE status before plant startup. 1E

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 3.0.4 do not apply because they would delay placing the facility in a lower MODE of operation.

ACTION times allow for an orderly sequential shutdown of both units when the inoperability of a component(s) affects both units with equal severity. If a shutdown is required for both units, a total of 18 hours is [illegible] for 6 to reach [illegible]. #1

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INSERT ATTACHED 3.0.6

3/4.0 APPLICABILITY

BASES

Specification 3.0.5 delineates the applicability of each specification to Unit ~~1~~ 3 and Unit ~~2~~ 4 operation. E

Specification 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10 CFR 50.36(c)(3):

"Surveillance requirements are requirements relating to test, calibration or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during the OPERATIONAL MODES or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a MODE or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an OPERATIONAL MODE for which the requirements of the associated Limiting Condition for operation do not apply unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception to the requirements of a specification.

Specification 4.0.2 establishes the conditions under which the specified time interval for Surveillance Requirements may be extended. Item a. permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. Item b. limits the use of the provisions of item a. to ensure that it is not used repeatedly to extend the surveillance interval beyond that specified. The limits of Specification 4.0.2 are based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. These provisions are sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

SPEC B3.0.~~5~~ 6 This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason. This ensures that one full division (train) and associated emergency power supply will be OPERABLE

The provisions of this specification permit the ACTION statements associated with individual systems, subsystems, trains, components or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source. #2

For example, Specification ~~3.7.1~~ 3.8.1.1 requires in part that two emergency diesel generators be OPERABLE. The ACTION statement provides for an out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.~~5~~6 all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.6 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components and devices must be OPERABLE, or otherwise satisfy Specification 3.0.6 (i.e., be capable of performing their design function and have at least one ~~normal or~~ one emergency power source OPERABLE). If they are not satisfied, shutdown is required. ~~in accordance with this specification~~.

In MODES 5 or 6, Specification 3.0.6 is not applicable and the electrical power specified in the definition of OPERABLE does not require emergency diesel capability beyond that required by Specification 3.8.1.2. The individual ACTION statements for each applicable Limiting Condition for operation in these MODES must be adhered to.

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3/4.0 APPLICABILITY

BASES

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be OPERABLE when Surveillance Requirements have been satisfactorily performed within the specified time interval. However, nothing in this provision is to be construed as implying that systems or components are OPERABLE when they are found or known to be inoperable although still meeting the Surveillance Requirements. This specification also clarifies that the ACTION requirements are applicable when Surveillance Requirements have not been completed within the allowed surveillance interval and that the time limits of the ACTION requirements apply from the point in time it is identified that a surveillance has not been performed and not at the time that the allowed surveillance interval was exceeded. Completion of the Surveillance Requirement within the allowable outage time limits of the ACTION requirements restores compliance with the requirements of Specification 4.0.3. However, this does not negate the fact that the failure to have performed the surveillance within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, was a violation of the OPERABILITY requirements of a Limiting Condition for Operation that is subject to enforcement action. Further, the failure to perform a surveillance within the provisions of Specification 4.0.2 is a violation of a Technical Specification requirement and is, therefore, a reportable event under the requirements of 10 CFR 50.73(a)(2)(i)(B) because it is a condition prohibited by the plant's Technical Specifications.

entering the ACTION statement.

If the allowable outage time limits of the ACTION requirements are less than 24 hours or a shutdown is required to comply with ACTION requirements, e.g., Specification 3.0.3, a 24-hour allowance is provided to permit a delay in ~~implementing the ACTION requirements.~~ This provides an adequate time limit to #1 complete Surveillance Requirements that have not been performed. The purpose of this allowance is to permit the completion of a surveillance before a shutdown is required to comply with ACTION requirements or before other remedial measures would be required that may preclude completion of a surveillance. The basis for this allowance includes consideration for plant conditions, adequate planning, availability of personnel, the time required to perform the surveillance, and the safety significance of the delay in completing the required surveillance. The provision also provides a time limit for the completion of Surveillance Requirements that become applicable as a consequence of MODE changes imposed by ACTION requirements and for completing Surveillance Requirements that are applicable when an exception to the requirements of Specification 4.0.4 is allowed. If a surveillance is not completed within the 24-hour allowance, the time limits of the ACTION requirements are applicable at that time. When a surveillance is performed within the 24-hour allowance and the Surveillance Requirements are not met, the time limits of the ACTION requirements are applicable at the time that the surveillance is terminated.

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Surveillance Requirements do not have to be performed on inoperable equipment because the ACTION requirements define the remedial measures that apply. However, the Surveillance Requirements have to be met to demonstrate that inoperable equipment has been restored to OPERABLE status.

Specification 4.0.4 establishes the requirement that all applicable surveillances must be met before entry into an OPERATIONAL MODE or other condition of operation specified in the Applicability statement. The purpose of this specification is to ensure that system and component OPERABILITY requirements or parameter limits are met before entry into a MODE or condition for which these systems and components ensure safe operation of the facility. This provision applies to changes in OPERATIONAL MODES or other specified conditions associated with plant shutdown as well as startup.

Under the provisions of this specification, the applicable Surveillance Requirements must be performed within the specified surveillance interval to ensure that the Limiting Conditions for Operation are met during initial plant startup or following a plant outage.

When a shutdown is required to comply with ACTION requirements, the provisions of Specification 4.0.4 do not apply because this would delay placing the facility in a lower MODE of operation.

Specification 4.0.5 establishes the requirement that inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. These requirements apply except when relief has been provided in writing by the Commission.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an OPERATIONAL MODE or other specified condition takes precedence over the ASME Boiler and Pressure

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3/4.0 APPLICABILITY

Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

Specification 4.0.6 delineates the applicability of the surveillance activities to Unit ~~1~~ 3 and Unit ~~2~~ 4 operations. E

Successful completion of a shared component surveillance may also be used to meet the surveillance requirement on the other unit. Unit-specific components that are part of a shared systems must meet their respective surveillance requirements (e.g., AFW motor operated valves and flow control valves). #1

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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. Figure 3.1-1 shows the SHUTDOWN MARGIN equivalent to 1.77% Δk/k at the end-of-core-life with respect to an uncontrolled cooldown. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With ~~T_{avg}~~ RCS Average Temperature less than 200°F, the reactivity transients resulting from an inadvertent ~~a postulated steam line break~~ cooldown of the RCS or an inadvertent dilution of RCS boron are minimal and a 1% Δk/k SHUTDOWN MARGIN provides adequate protection. #1

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 -3.5×10^{-4} Δk/k/°F. The MTC value of -3.0×10^{-4} Δk/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 -3.5×10^{-4} Δk/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°F. This limitation is required to ensure: (1) the moderator temperature coefficient is within it analyzed temperature range, (2) the trip instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor 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. One flow path from the charging pump discharge is acceptable since the flow path components subject to an active failure are upstream of the charging pumps.

BORATION SYSTEMS (Continued)

The boration flow path specification allows the RWST and the boric acid storage tank to be the boron sources. Due to the lower boron concentration in the RWST, borating the RCS from this source is less effective than borating from the boric acid tank and additional time may be required to achieve the desired SHUTDOWN MARGIN required by ACTION statement restrictions.

The ACTION statement restrictions for the boration flow paths allow continued operation in mode 1 for a limited time period with either boration source flow path or the normal flow path to the RCS (via the regenerative heat exchanger) inoperable. In this case, the plant capability to borate and charge into the RCS is limited and the potential operational impact of this limitation on mode 1 operation must be addressed. With both the flow path from the boric acid tanks and the regenerative heat exchanger flow path inoperable, immediate initiation of action to go to COLD SHUTDOWN is required but no time is specified for the mode reduction due to the reduced plant capability with these flow paths inoperable.

Two charging pumps with independent power supplies are required to be OPERABLE to ensure single functional capability in the event an assumed failure renders one of the pumps or power supplies inoperable. However, the ACTION statement restrictions allow 7 days to restore an inoperable pump provided that two charging pumps are available. This restriction is acceptable based on the low probability of losing the power source common to both charging pumps. The bus supplying the pumps can be fed from either the Emergency Diesel Generator or the offsite grid through the startup transformer.

The boration capability of either flow path is sufficient to provide the required SHUTDOWN MARGIN in accordance with Figure 3.1-1 from expected operating conditions 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 3080 gallons of 20,000 PPM borated water from the boric acid storage tanks or 320,000 gallons of 1950 PPM borated water from the refueling water storage tank (RWST).

With the RCS temperature below 200°F, one boron injection source flow path 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 boron injection system source flow path becomes inoperable.

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 500 gallons of 20,000 ppm borated water from the boric acid storage tanks or 20,000 gallons of 1950 ppm borated water from the RWST.

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BORATION SYSTEMS (Continued)

The charging pumps are demonstrated to be OPERABLE by testing as required by Section XI of the ASME code or by specific surveillance requirements in the specification. These requirements are adequate to determine OPERABILITY because no safety analysis assumption relating to the charging pump performance is more restrictive than these acceptance criteria for the pumps.

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 temperature requirements for the RWST are based on the containment integrity LOCA analysis assumptions #2

The OPERABILITY of one Boron Injection System during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The OPERABILITY of the redundant heat tracing channels associated with the boric acid tank system ensures that the solubility of the boron solution will be maintained above the solubility limit of 135°F at 22,500 ppm boron.

One channel of heat tracing is sufficient to maintain the specified temperature limit. Since one channel of heat tracing is sufficient to maintain the specified temperature, operation with one channel out-of-service is permitted for a period of 30 days provided additional temperature surveillance is performed.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that: (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of rod misalignment on associated accident analyses are limited. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits continue. OPERABLE condition for the analog rod position indicators is defined as being capable of indicating rod position to within ±12 steps of the ~~REFERENCE POSITION.~~ demand counter position For the Shutdown Banks and Control Banks A and B, the ~~REFERENCE POSITION~~ Position Indication Requirement is defined as the group demand counter indicated position between 0 and 30 ~~steps withdrawn~~ inclusive, and between 200 and 228 steps withdrawn inclusive. This permits the operator to verify that the control rods in these banks are either fully withdrawn or fully inserted, the normal operating modes for these banks. Knowledge of these bank positions in these two areas satisfies all accident analysis assumptions concerning their position. For Control Banks C and D, the ~~REFERENCE POSITION~~ Position Indication Requirement is defined as the group demand counter indicated position between 0 and ~~30 steps withdrawn inclusive, and between 150 and~~ 228 steps withdrawn inclusive. ~~For the withdrawal range of 31 to 149 steps inclusive the REFERENCE POSITION is defined as the individual rod calibration curve noting indicated analog rod position vs indicated group demand counter~~ #1

MOVABLE CONTROL ASSEMBLIES (Continued)

~~position. Comparison of the indicated analog rod position to the calibration curve is sufficient to allow determination that a control rod is indeed misaligned from its bank.~~ Comparison of the group demand counters to the bank insertion limits with verification of rod position with the analog rod position indicators (after thermal soak after rod motion) is sufficient verification that the control rods are above the insertion limits. #1

Rod position indication is provided by two methods: a digital count of actuating pulses which shows demand position of the banks and a linear position indicator Linear Variable Differential Transformer which indicates the actual rod position. The relative accuracy of the linear position indicator Linear Variable Differential Transformer is such that, with the most adverse error, an alarm will be actuated if any two rods within a bank deviate by more than 24 steps for rods in motion and 12 steps for rods at rest. Complete rod misalignment (12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at RATED THERMAL POWER.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors and a restriction in THERMAL POWER. These restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 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 LCOs are satisfied.

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to 1.30 during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ 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; and

$F_{xy}(Z)$ 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

Replace 3/4.2.1 FPL WORDING

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

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Replace 3/4.2.1
FPL WORDING

#1

AXIAL FLUX DIFFERENCE (Continued)

Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 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.

PLANT PROCESS COMPUTER

Provisions for monitoring the AFD on an automatic basis are derived from the ~~Digital Data Process System~~ through the alarm used to monitor the AFD. The ~~Digital Data Process System determines the 1-minute average of each of~~ MONITORS the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two 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.

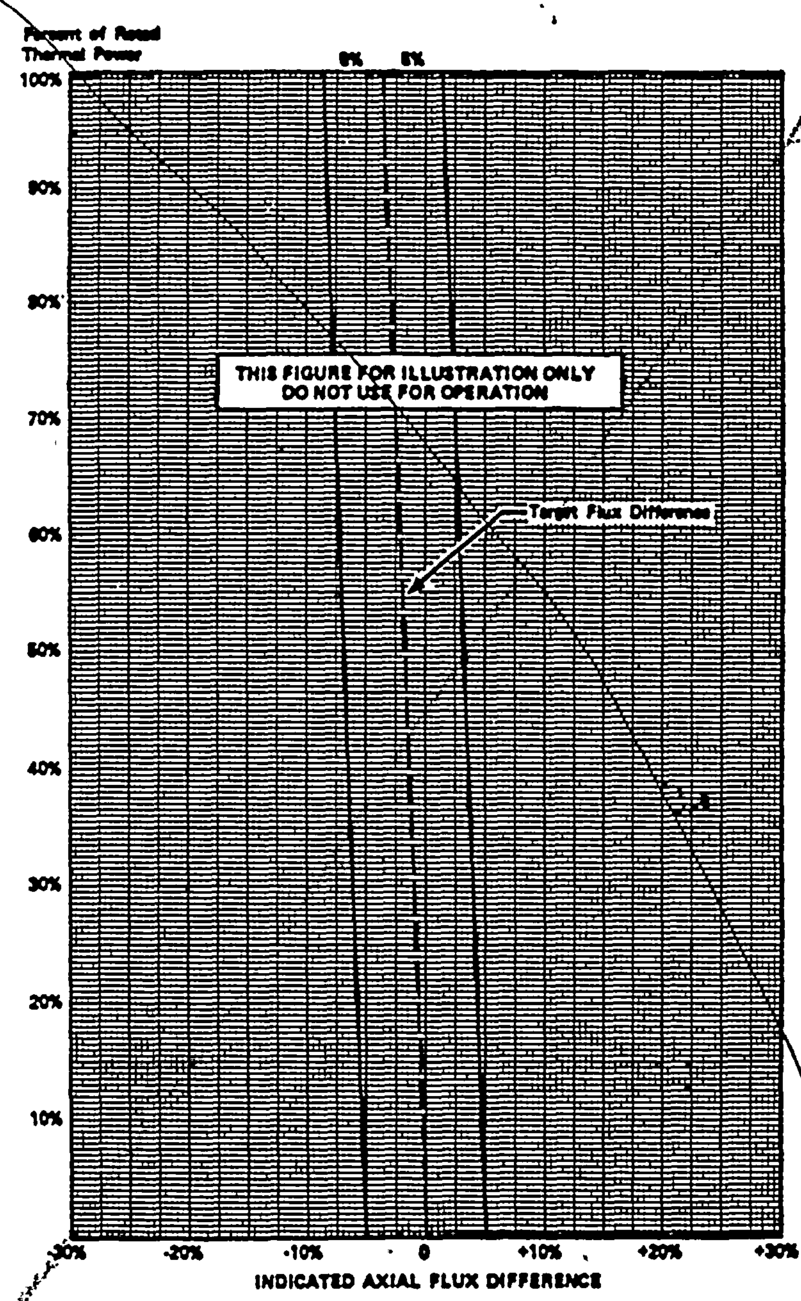

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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Replace 3/4.2.2 and 3/4.2.3 with FPL WORDING

#1

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained. The relaxation of $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

Replace 3/4.2.2 and 3/4.2.3 with FPL WORDING #1

HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

Fuel rod bowing reduces the value of DNB ratio. Credit is available to offset this reduction in the generic margin. The generic margins, totaling 9.1% DNBR completely offset any rod bow penalties. This margin includes the following.

a. Design limit DNBR of [1.30 vs 1.28],

b. Grid Spacing (K_s) of [0.046 vs 0.059],

c. Thermal Diffusion Coefficient of [0.038 vs 0.059],

d. DNBR Multiplier of [0.86 vs 0.88], and

e. Pitch reduction.

The applicable values of rod bow penalties are referenced in the FSAR.

When an F_Q measurement is taken, an allowance for both experimental error and manufacturing tolerance must be made. An allowance of 5% is appropriate for a full-core map taken with the Incore Detector Flux Mapping System, and a 3% allowance is appropriate for manufacturing tolerance.

The Radial Peaking Factor, $F_{xy}(Z)$, is measured periodically to provide assurance that the Hot Channel Factor, $F_Q(Z)$, remains within its limit. The F_{xy} limit for RATED THERMAL POWER (F_{xy}^{RTP}) as provided in the Radial Peaking Factor Limit Report per Specification 6.9.1.6 was determined from expected power control manuevers over the full range of burnup conditions in the core.

When RCS flow rate and $F_{\Delta H}^N$ are measured, no additional allowances are necessary prior to comparison with the limits of Figures 3.2-3 and 3.2-4. Measurement errors of 2.1% for RCS total flow rate and 4% for $F_{\Delta H}^N$ have been allowed for in determination of the design DNBR value.

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 limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

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QUADRANT POWER TILT RATIO (Continued)

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action action does not correct the tilt, the margin for uncertainty on $F_Q(Z)$ is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or incore thermocouple map are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8. #2

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 above the applicable design limits throughout each analyzed transient. The indicated T_{avg} value of 576.3°F and the indicated pressurizer pressure value of ~~2220~~ 2222 psig correspond to analytical limits of 578.2°F and ~~2220~~ 2205 psig respectively, with allowance for measurement uncertainty. (E)

The indicated RCS flow value of 277,900 gpm corresponds to an analytical limit of 268,500 gpm which is assumed to have a 3.5% measurement uncertainty. The above measurement uncertainty estimates assume that these instrument channel outputs are averaged to minimize the uncertainty.

The 12-hour periodic surveillance of Tavg and pressurizer pressure and the refueling surveillance for RCS flow are ~~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. (E)

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Strict control of the flux difference (and rod position) is not as necessary during operation at less than 90% power. This is because xenon distribution control is not as significant as the control at full power and allowance has been made in predicting the heat flux peaking factors for less strict control at less than 90% power. Although it is intended that the plant will be operated with the AFD within the target band required by Specification 3.2.1 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. If the flux difference exceeds the limit for a cumulative period of one hour in any 24 hours, then xenon distributions may be significantly changed and hence operation at less than 50% power is required to protect against potentially more severe consequences of some accidents. For THERMAL POWER levels ~~below 50%~~ 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.

Strict control of flux differences is not possible during certain physics tests or during the required periodic excore calibrations. Therefore, this specification is not applicable during physics testing or excore calibration provided the duration of the deviation is limited. This is acceptable due to the extremely low probability of a significant accident during these operations. The time outside the target band is limited to 16 hours.

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AXIAL FLUX DIFFERENCE (Continued)

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer ~~determines~~ monitors ~~the 1-minute average of each of~~ the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for two 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 ~~below~~ 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 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor 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 degrees-F ECCS acceptance criteria limit. The LOCA peak fuel clad temperature limit may be sensitive to the number of steam generator tubes plugged. The current limit is valid for tube plugging levels up to 5%.

$F_Q(Z)$, 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.

$F^N_{\Delta H}$, 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.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6.

SAME FIGURE AS P&R

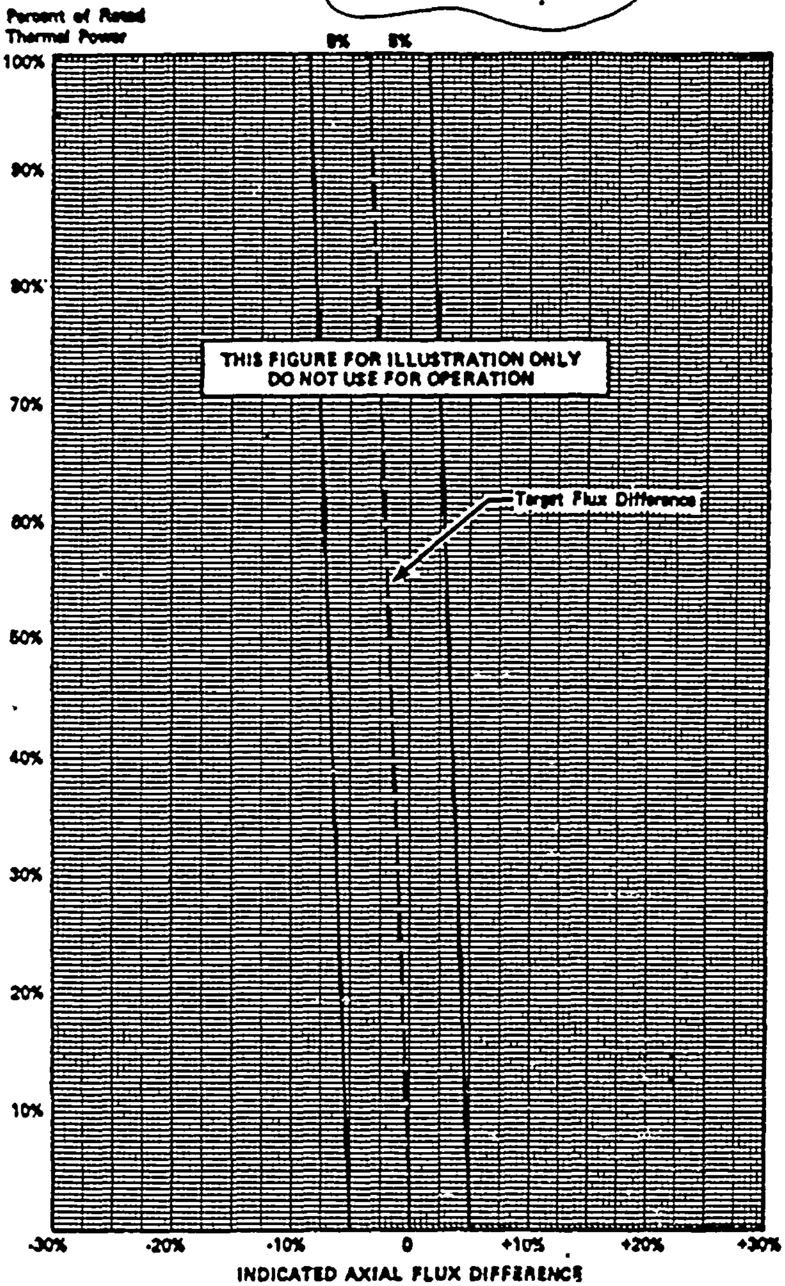

FIGURE B 3/4 2-1

TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER

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HEAT FLUX HOT CHANNEL FACTOR, 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; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. These uncertainties only apply if the map is taken for purposes other than the determination of P_{BL} and P_{RB}.

$F^N_{\Delta H}$ will be maintained within its limits provided Conditions a. through d. above are maintained.

In the specified limit of $F^N_{\Delta H}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F^N_{\Delta H} \leq 1.62/1.08$. The logic behind the larger uncertainty in this case is that (a) normal perturbations in the radial power shape (e.g., rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q, (b) although the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$ and (c) an error in the prediction for radial power shape, which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control, but compensation for $F_{\Delta H}$ is less readily available. When a measurement of $F^N_{\Delta H}$ is taken, experimental error must be allowed for and 4% is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system.

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HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The following are independent augmented surveillance methods used to ensure peaking factors are acceptable for continued operation above Threshold Power, P_T:

Base Load - This method uses the following equation to determine peaking factors:

$$F_{QBL} = F_Q(z) \text{ measured} \times 1.09 \times W(z)_{BL}$$

where: $W(z)_{BL}$ accounts for power shapes
1.09 accounts for uncertainty
$F_Q(z)$ measured data
F_{QBL} = Base load peaking factor

The analytically determined $[F_Q]^P$ is formulated to generate limiting shapes for all load follow maneuvers consistent with control to a ± 5% band about the target flux difference. For Base Load operation the severity of the shapes that need to be considered is significantly reduced relative to load follow operation.

The severity of possible shapes is small due to the restrictions imposed by Sections 4.2.2.3. To quantify the effect of the limiting transients which could occur during Base Load operation, the function $W(Z)_{BL}$ is calculated from the following relationship:

$$W(Z)_{BL} = \text{Max}\left[\frac{F_Q(Z)\text{(Base Load Case(s), 150 MWD/T)}}{F_Q(Z)\text{ (ARO, 150 MWD/T)}}, \frac{F_Q(Z)\text{(Base Case(s), 85\% EOL BU)}}{F_Q(Z)\text{ (ARO, 85\% BOL BU)}}\right]$$

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Radial Burndown - This method uses the following equation to determine peaking factors:

$$F_Q(z)_{R.B.} = F_{xy}(z)_{measured} \times F_z(z) \times 1.09$$

where: 1.09 accounts for uncertainty
$F_z(z)$ accounts for axial power shapes
$F_{xy}(z)$ measured = ratio of peak power density to average power density at elevation (z).
$F_{Q\,R.B.}$ = Radial Burndown Peaking Factor

For Radial Burndown operation the full spectrum of possible shapes consistent with control to a ± 5% Delta-I band needs to be considered in determining power capability. Accordingly, to quantify the effect of the limiting transients which could occur during Radial Burndown operation, the function $F_z(Z)$ is calculated from the following relationship:

$F_z(Z) = [F_Q(Z)]$ FAC Analysis/$[F_{xy}(Z)]$ ARO

The essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This can be accomplished by using the boron system to position the full length control rods to produce the required indicated flux difference.

Above the power level of P_T, additional flux shape monitoring is required. In order to assure that the total power peaking factor, F_Q, is maintained at or below the limiting value, the movable incore instrumentation will be utilized. Thimbles are selected initially during startup physics tests so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor F_Q can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived from incore measurements, i.e., an effective radial peaking factor R, can be determined as the ratio of the total peaking factor resulting from a full core flux map and the axial peaking factor in a selected thimble.

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HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

The limiting value of $[F_j(Z)]_s$ is derived as follows:

$$[F_j(Z)]_s = \frac{[F_Q]^L \times [K(Z)]}{P_L \bar{R}_j (1+\sigma_j)(1.03)(1.07)}$$

Where:

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

b) P_L is reactor thermal power expressed as a fraction of 1.

c) K (Z) is the reduction in the F_Q limit as a function of core elevation (Z) as determined from Figure 3.2-2.

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

e) $\bar{R}_j$, for thimble j, is determined from n=6 incore flux maps covering the full configuration of permissible rod patterns at the thermal power limit of P_T.

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

where

$$R_{ij} = \frac{F_{Qi}\ \text{meas.}}{[F_{ij}(Z)]\ \max}$$

and $F_{ij}(Z)$ is the normalized axial distribution at elevation Z from thimble j in map i which has a measured peaking factor without uncertainties or densification allowance of F_{Qi} meas.

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

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

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HEAT FLUX HOT CHANNEL FACTOR, AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

g) The factor 1.03 reduction in the kw/ft limit is the engineering uncertainty factor.

h) The factors $(1+\sigma_j)$ and 1.07 represent the margin between $(F_j(Z)]_L$ limit and the MIDS alarm setpoint $[F_j(Z)]_S$. Since $(1+\sigma_j)$ is bounded by a lower limit of 1.02, there is at least a 9% reduction of the alarm setpoint. Operations are permitted in excess of the operational limit $\leq$ 4% while making power adjustment on a percent for percent basis.

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To accommodate instrument drift that may occur between operational tests, the actual trip setpoint is normally set at a more conservative limit than the setpoint allowed by Table 3.3-3

3/4.3.1 and 3/4.3.2 REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Trip System and the Engineered Safety Features Actuation System instrumentation and interlocks ensures that: (1) the associated ACTION and/or Reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its Setpoint (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out-of-service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses. The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The Engineered Safety Features Actuation System limiting Instrumentation Trip Setpoints specified in Table 3.3-3 are the ~~nominal~~ values at which the trips are set for each functional unit. A Setpoint is considered to be adjusted consistent with the ~~nominal~~ value when the "as measured" Setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which Setpoints can be measured and calibrated, Allowable Values for ~~the Setpoints have~~ been specified in Table 3.3-3. Operation with Setpoints ~~less conservative than the~~ Trip Setpoint but within the Allowable Value is acceptable since an allowance ~~has been made~~ in the transient safety analysis to accommodate this error.

The methodology to derive the Trip ~~Setpoints is based upon combining all of the uncertainties~~ includes an allowance for instrument uncertainties in the channels. Inherent to the determination of the Trip Setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. ~~Rack drift in excess of the~~ Allowable Value exhibits the behavior ~~that~~ the rack has not met its allowance. ~~Being that there is a small~~ statistical chance that this will happen, an ~~infrequent~~ excessive ~~drift~~ is expected. Rack or sensor drift, in excess ~~of~~ the allowance that is more than ~~occasional~~, may be indicative of ~~more~~ serious problems and should warrant further investigation. #1

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents events, and transients. Once the required logic combination is completed, the system sends actuation signals to

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REACTOR TRIP SYSTEM and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION (Continued)

those Engineered Safety Features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss-of-coolant accident: (1) Safety Injection pumps start and automatic valves position, (2) Reactor trip, (3) feed water isolation, (4) startup of the emergency diesel generators, (5) 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.)E

intake cooling water and component cooling

The Engineered Safety Features Actuation System interlocks perform the following functions:

HIGH STEAM FLOW SAFETY INJECTION BLOCK - This permissive is used to block the safety injection (SI) signal generated by High Steam Line Flow coincident with Low Steam Line Pressure or Low T_{avg}. The permissive is generated when two out of three Low T_{avg} channels drop below their setpoints and the manual SI Block/Unblock switch is momentarily placed in the block position. This switch is a spring return to the normal position type. The permissive will automatically be defeated if two out of three Low T_{avg} channels rise above their setpoints. The permissive may be manually defeated when two out of three Low T_{avg} channels are below their setpoints and the manual SI Block/Unblock switch is momentarily placed in the unblock position.

LOW PRESSURIZER PRESSURE SAFETY INJECTION BLOCK - This permissive is used to block the safety injection signals generated by Low Pressurizer Pressure and High Differential Pressure between the Steam Line Header and any Steam Line. The permissive is generated when two out of three pressurizer pressure permissive channels drop below their setpoints and the manual SI Block/Unblock switch is momentarily placed in the block position. This is the same switch that is used to manually block the High Steam Flow Safety Injection signals mentioned above. This permissive will automatically be defeated if two out of three pressurizer pressure permissive channels rise above their setpoints. The permissive channels rise above their setpoints. The permissive may be manually defeated when two out of three pressurizer pressure permissive channels are below their setpoints and the manual SI Block/Unblock switch momentarily placed in the Unblock position.

~~PARTIAL AUXILIARY FEEDWATER START - Bus Stripping on Bus A opens MOV-1404 and MOV-1405 and Bus Stripping on Bus B opens MOV-1403 and MOV-1404 (See Specification 3/4.7.1.2, Table 3.7-3).~~ #13

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING FOR PLANT OPERATIONS

The OPERABILITY of the radiation monitoring instrumentation for plant operations ensures that conditions indicative of potential uncontrolled radioactive releases are monitored and that appropriate actions will be automatically or manually initiated when the radiation level monitored by each channel reaches its alarm or trip setpoint.

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 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.

3/4.3.3.3 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, Revision 3, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," May 1983 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.4 FIRE DETECTION INSTRUMENTATION

The OPERABILITY of the fire detection instrumentation ensures that both adequate warning capability is available for prompt detection of fires and that Fire Suppression Systems, that are actuated by fire detectors, will discharge extinguishing agents in a timely manner. Prompt detection and suppression of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility Fire Protection Program.

Fire detectors that are used to actuate Fire Suppression Systems represent a more critically important component of a plant Fire Protection Program than detectors that are installed solely for early fire warning and notification. Consequently, the minimum number of OPERABLE fire detectors must be greater.

The loss of detection capability for Fire Suppression Systems, actuated by fire detectors, represents a significant degradation of fire protection for

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FIRE DETECTION INSTRUMENTATION (Continued)

any area. As a result, the establishment of a fire watch patrol must be initiated at an earlier stage than would be warranted for the loss of detectors that provide only early fire warning. The establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.5 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The Alarm/Trip Setpoints for these instruments shall be calculated and adjusted in accordance with the methodology and parameters 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.

3/4.3.3.6 RADIOACTIVE GASEOUS EFFLUENT MONITORING 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 and adjusted in accordance with the methodology and parameters 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 GAS DECAY TANK 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. The sensitivity of any noble gas activity monitors used to show compliance with the gaseous effluent release requirements of Specification 3.11.2.2 shall be such that concentrations as low as 1×10^{-6} µCi/ml are measurable.

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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 the applicable design limit 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 6 hours.

In MODE 3, three reactor coolant loops provide sufficient heat removal capability for removing core decay heat in the event of a bank withdrawal accident; however, a single reactor coolant loop provides sufficient heat removal capacity if a bank withdrawal accident can be prevented, i.e., by opening the Reactor Trip ~~System~~ breakers or by placing the Rod Control System in the Bank Select Mode with a Shutdown Bank Selected. ~~Single active failure considerations require that at least two loops be OPERABLE at all times with the Reactor Trip System breakers open or the Rod Control System in the Bank Select Mode with a Shutdown Bank Selected.~~ #2

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 active failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.~~ #2

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 (RCP) or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting an RCP with one or more RCS cold legs less than or equal to 275°F are provided to prevent RCS pressure transients, caused by energy additions from the Secondary Coolant 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 reactor coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each ~~steam generator is less than 50°F above each~~ of the RCS cold leg temperatures. The 50°F limit includes instrument error. #1

The Technical Specifications for Hot Shutdown and Cold Shutdown allow an inoperable RHR pump to be the operating RHR pump for up to 2 hours for surveillance testing to establish operability. This is required because of the piping arrangement when the RHR system is being used for Decay Heat Removal.

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3/4.4.2 SAFETY VALVES 293,300

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 ~~280,000~~ 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 RCS vent opening of at least 2.20 square inches will provide overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure ~~Protection~~ Mitigating System provides a diverse means of protection against RCS overpressurization at low temperatures. #1 E E

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 Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The 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 FSAR. The limit is consistent with the initial FSAR 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 (~~1196~~ 1133 cubic feet) also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that both backup pressurizer heater groups be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation. #1

BASES

3/4.4.4 ~~RELIEF~~ PORV BLOCK 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.~~ The opening of the PORVs fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2 or 3. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. #1

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the Reactor Coolant System and the Secondary Coolant System (reactor-to-secondary leakage = 500 gallons per day ~~per~~ through any one steam generator). E Cracks having a reactor-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that reactor-to-secondary leakage of 500 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with the all volatile treatment (AVT) 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.

PR

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 in a Special Report pursuant to Specification 6.9.2 within 30 days and prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

At least one channel of

The RCS Leakage Detection Systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary to the containment. The containment sump level system is the normal sump level instrumentation. The Post Accident Containment Water Level Monitor - Narrow range instrumentation also satisfies the requirement for a sump level monitoring system. In addition, gross leakage will be detected by changes in makeup water requirements, visual inspection, and audible detection. Leakage to other systems will be detected by activity changes (e.g., within the component cooling system) or water inventory changes (e.g., tank levels). E

3/4.4.6.2 OPERATIONAL LEAKAGE — per Table 3.3-5

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 dose guideline values in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 500 gpd leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

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 limitations restrict operation when the flow from the reactor coolant pump seals exceeds 30 gpm at a nominal RCS pressure of 2235 psig. #7

OPERATIONAL LEAKAGE (Continued)

The leakage from any RCS pressure isolation valve is sufficiently low to ensure early detection of possible in-series valve failure. It is apparent that when pressure isolation is provided by two in-series valves and when failure of one valve in the pair can go undetected for a substantial length of time, verification of valve integrity is required. Since these valves are important in preventing overpressurization and rupture of the ECCS low pressure piping which could result in a LOCA, these valves should be tested periodically to ensure low probability of gross failure.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. ~~Leakage from the RCS pressure isolation valve is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.~~ The leakage identified from these surveillances shall not be considered as IDENTIFIED LEAKAGE unless they result in actual mass loss from the RCS in which case the leakage shall be IDENTIFIED LEAKAGE. #8

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 reactor coolant ensure that the resulting 2-hour doses at the SITE BOUNDARY will not exceed an

PR

SPECIFIC ACTIVITY (Continued)

appropriately small fraction of 10 CFR Part 100 dose guideline values following a steam generator tube rupture accident in conjunction with an assumed steady-state reactor-to-secondary steam generator leakage rate of 1 gpm. The values 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 Turkey Point site, Units 3 and 4 site, such as SITE BOUNDARY location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the reactor coolant's specific activity greater than 1 microCurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

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The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low reactor coolant limit of 1 microCurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the reactor coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of reactor coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than ~~10~~ minutes from these determinations has been made for several reasons. The first consideration is the difficulty to identify short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the reactor coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of ~~10~~ 30 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical reactor coolant radioactivity. ~~The radionuclides in the typical reactor coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before they could be transported from the reactor coolant to the SITE BOUNDARY under any accident condition.~~ #1

#1

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PR

SPECIFIC ACTIVITY (Continued)

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. After 90 minutes, the gross count should be made in a reproducible geometry of sample and counter having reproducible beta or gamma self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides. The radiochemical determination of nuclides should be based on multiple counting of the sample within typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing ~~T_{avg}~~ average coolant temperature to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the reactor coolant is below the lift pressure of the atmospheric steam relief valves. The Surveillance Requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained. E

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the RCS are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. During RCS heatup and cooldown, the temperature and pressure changes must be limited to be consistent with design assumptions and to satisfy stress limits for brittle fracture.

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 to 3.4-5 for the service period specified thereon:

    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; and

    b. Figures 3.4-2 to 3.4-5 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.

PRESSURE/TEMPERATURE LIMITS (Continued)

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/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 300°F, and

5. System preservice hydrotests and inservice 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.

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 10 effective full power years (EFPY) of service life. The 10 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in the core region is greater than the RT_{NDT} of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Tables B 3/4.4-1 and B 3/4.4-2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference tempera ture, based upon the fluence and chemistry content of the material has been predicted using Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 to 3.4-5 include predicted adjustments for this shift in RT_{NDT} at the end of 10 EFPY as well as adjustments for possible errors in the pressure and temperature sensing instruments.

TABLE B 3/4.4-1
REACTOR VESSEL TOUGHNESS DATA
TURKEY POINT - UNIT 3

Component	Material Type	Cu (%)	P (%)	NDTT (°F)	50 ft lb/35 mils Lateral Expansion Temp (°F) Long	Trans	RT_{NDT} (°F)	Minimum Upper Shelf (ft lb) Long	Trans
Cl. Hd. Dome	A302 Gr. B	-	-	0	-	36(a)	0	> 70	> 45.5(a)
Cl. Hd. Flange	A508 Cl. 2	-	-	44(a)	-	31(a)	44	>118	> 76.5(a)
Ves. Sh. Flange	A508 Cl. 2	-	-	-23(a)	-	-41(a)	-23	>120	> 78(a)
Inlet Nozzle	A508 Cl. 2	-	-	60(a)	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	-	60(a)	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	-	60(a)	-	NA	60	NA	NA
Outlet Nozzle	A508 Cl. 2	-	-	27(a)	-	9(a)	27	>110	>71.5(a)
Outlet Nozzle	A508 Cl. 2	-	-	7(a)	-	-22(a)	7	>111	>72(a)
Outlet Nozzle	A508 Cl. 2	-	-	42(a)	-	23(a)	42	>140	>91(a)
Upper Shell	A508 Cl. 2	-	-	50	-	44(a)	50	>129	>83.5(a)
Inter. Shell	A508 Cl. 2	0.058	0.010	40	-	25(a)	40	>122	>79(a)
Lower Shell	A508 Cl. 2	0.079	0.010	30	-	2(a)	30	163	106(a)
Trans. Ring	A508 Cl. 2	-	-	60(a)	-	58(a)	60	>109	>70.5(a)
Bot. Hd. Dome	A302 Gr. B	-	-	-10	-	NA	30(a)	NA	NA
Inter. to Lower Shell Girth Weld	SAW	0.31	0.011	0(a)	-	63	3	-	63
HAZ	HAZ	-	-	0(a)	-	0	0	-	168

(a) Estimated values based on NUREG-0800, Branch Technical Position - MTEB 52

TABLE B 3/4.4-2
REACTOR VESSEL TOUGHNESS DATA
TURKEY POINT - UNIT 4

Component	Material Type	Cu (%)	P (%)	NDTT (°F)	50 ft lb/35 mils Lateral Expansion Temp (°F)		RT_{NDT} (°F)	Minimum Upper Shelf (ft lb)	
					Long	Trans		Long	Trans
Cl. Hd. Dome	A302 Gr. B	-	-	-20	-	NA	30	NA	NA
Cl. Hd. Flange	A508 Cl. 2	-	-	-4[a]	-	27[a]	-4	199	129[a]
Ves. Sh. Flange	A508 Cl. 2	-	-	-1[a]	-	-11[a]	-1	176	114[a]
Inlet Nozzle	A508 Cl. 2	-	-	60[a]	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	-	60[a]	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	-	16[a]	-	13[a]	16	162	105[a]
Outlet Nozzle	A508 Cl. 2	-	-	7[a]	-	-25[a]	7	165	107[a]
Outlet Nozzle	A508 Cl. 2	-	-	38[a]	-	16[a]	38	160	104[a]
Outlet Nozzle	A508 Cl. 2	-	-	60[a]	-	42[a]	60	143	93[a]
Upper Shell	A508 Cl. 2	-	-	40	-	32[a]	40	156	101[a]
Inter. Shell	A508 Cl. 2	0.054	0.010	50	-	90[a]	50	143	93[a]
Lower Shell	A508 Cl. 2	0.056	0.010	40	-	38[a]	40	149	97[a]
Trans. Ring	A508 Cl. 2	-	-	60[a]	-	30[a]	60	NA	NA
Bot. Hd. Dome	A302 Gr. B	-	-	10	-	30[a]	10	NA	NA
Inter. to Lower Shell Girth Weld	SAW	0.31	0.011	0[a]	-	63	3	NA	63
HAZ	HAZ	-	-	0	-	NA	0	NA	140

(a) Estimated values b[illegible] on NUREG-0800, Bra[illegible]hnical Position - MTEB [illegible]

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PRESSURE/TEMPERATURE LIMITS (Continued)

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 Part 50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The lead factor represents the relationship between the fast neutron flux density at the location of the capsule and the inner wall of the reactor vessel. Therefore, the results obtained from the surveillance specimens can be used to predict future radiation damage to the reactor vessel material by using the lead factor and the withdrawal time of the capsule. 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. The surveillance capsule "T" results from Unit 3 (WCAP 8631) and Unit 4 (SWRI 02-4221) were used to generate the heatup and cooldown curves in Figure 3.4-2 through 3.4-5.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50, and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical 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 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 nonductile 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:

PR

PRESSURE/TEMPERATURE LIMITS (Continued)

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\ K_{IM} + K_{It} \leq K_{IR} \quad (2)$$

Where: K_{IM} = the stress intensity factor caused by membrane (pressure) stress,

K_{It} = the stress intensity factor caused by the thermal gradients,

K_{IR} = constant provided by the Code as a function of temperature relative to the RT_{NDT} of the material,

C = 2.0 for level A and B service limits, and

C = 1.5 for inservice hydrostatic and leak test operations.

At any time during the heatup or cooldown transient, K_{IR} is determined by the metal temperature at the tip of the postulated flaw, the appropriate value for RT_{NDT}, and the reference fracture toughness curve. The thermal stresses resulting from temperature gradients through the vessel wall are calculated and then the corresponding thermal stress intensity 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 ΔT developed during cooldown results in a higher value of K_{IR} at the 1/4T location

PRESSURE/TEMPERATURE LIMITS (Continued)

for finite cooldown rates than for steady-state operation. Furthermore, if conditions exist such that the increase in K_{IR} exceeds K_{It}, the calculated allowable pressure during cooldown will be greater than the steady-state value.

The above procedures are needed because there is no direct control on temperature at the 1/4T location; therefore, allowable pressures may unknowingly be violated if the rate of cooling is decreased at various intervals along a cooldown ramp. The use of the composite curve eliminates this problem and assures conservative operation of the system for the entire cooldown period.

HEATUP

Three separate calculations are required to determine the limit curves for finite heatup rates. As is done in the cooldown analysis, allowable pressure-temperature relationships are developed for steady-state conditions as well as finite heatup rate conditions assuming the presence of a 1/4T defect at the inside of the vessel wall. The thermal gradients during heatup produce compressive stresses at the inside of the wall that alleviate the tensile stresses produced by internal pressure. The metal temperature at the crack tip lags the coolant temperature; therefore, the K_{IR} for the 1/4T crack during heatup is lower than the K_{IR} for the 1/4T crack during steady-state conditions at the same coolant temperature. During heatup, especially at the end of the transient, conditions may exist such that the effects of compressive thermal stresses and different K_{IR}'s for steady-state and finite heatup rates do not offset each other and the pressure-temperature curve based on steady-state conditions no longer represents a lower bound of all similar curves for finite heatup rates when the 1/4T flaw is considered. Therefore, both cases have to be analyzed in order to assure that at any coolant temperature the lower value of the allowable pressure calculated for steady-state and finite heatup rates is obtained.

The second portion of the heatup analysis concerns the calculation of 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.

PR

PRESSURE/TEMPERATURE LIMITS (Continued)

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 new 10 CFR 50 Appendix G rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. The rule states that the minimum metal temperature for the flange regions should be at least 120 F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig). Since the limiting RT_{NDT} for the flange regions for Turkey Point Units 3 and 4 is 44 F, the minimum temperature required for pressure of 621 psig and greater based on the Appendix G rule is 164 F. The heatup and cooldown curves as shown in Figures 3.4-2 to 3.4-5 clearly satisfy the above requirement by ample margins.

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 nonductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

~~LOW TEMPERATURE~~ OVERPRESSURE ~~PROTECTION~~ MITIGATING SYSTEM

The Technical Specifications provide requirements to isolate High Pressure Safety Injection from the RCS ~~by specifying the closure of MOV-843A, 843B, and 869 and the removal of their power supplies,~~ and to prevent the start of an idle RCP if secondary temperature is more than 50 F above the RCS cold leg temperatures. These requirements are designed to ensure that mass and heat input transients more severe than those assumed in the low temperature overpressurization protection analysis cannot occur. E

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.20 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 50°F above the RCS cold leg temperatures including margin for instrument error, or (2) the start of a HPSI pump and its injection into a water-solid RCS.

PR

PRESSURE/TEMPERATURE LIMITS (Continued)

REACTOR MATERIAL SURVEILLANCE PROGRAM

Each Type I capsule contains 28 V-notch specimens, ten Charpy specimens machined from each of the two shell forgings. The remaining eight Charpy specimens are machined from correlated monitor material. In addition, each Type I capsule contains four tensile specimens (two specimens from each of the two shell forgings). Dosimeters of copper, nickel, aluminum-cobalt, and cadmium-shielded aluminum-cobalt wire are secured in holes drilled in spacers at the top, middle and bottom of each Type I capsule.

Each Type II capsule contains 32 Charpy V-notch specimens: eight specimens machined from one of the shell forgings, eight specimens of weld metal and eight specimens of HAZ metal, the remaining eight specimens are correlation monitors. In addition, each Type II capsule contains four tensile specimens and four WOL forgings and the weld metal. Each Type II capsule contains a dosimeter block at the center of the capsule. Two cadmium-oxide-shielded capsules, containing the two isotopes uranium-238 and neptunium-237, are contained in the dosimeter block. The double containment afforded by the dosimeter assembly prevents loss and contamination by the neptunium-237 and uranium-238 and their activation products. Each dosimeter block contains approximately 20 milligrams of neptunium-237 and 13 milligrams of uranium-238 contained in a 3/8-inch OD sealed brass tube. Each tube is placed in a 1/2-inch diameter hole in the dosimeter block (one neptunium-237 and one uranium-238 tube per block), and the space around the tube is filled with cadmium oxide. After placement of this material, each hole is blocked with two 1/16-inch aluminum spacer discs and an outer 1/8-inch steel cover disc, which is welded in place. Dosimeters of copper, nickel, aluminum-cobalt and cadmium-shielded aluminum-cobalt are also secured in holes drilled in spacers located at the top, middle and bottom of each Type II capsule.

Capsule Type	Capsule Identification
I	S
II	V
II	T
I	U
II	X
I	W
I	Y
I	Z

PR

REACTOR COOLANT SYSTEM

BASES

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 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i).

Components of the Reactor Coolant System were designed to provide access ~~to permit~~ that originally allowed inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, ~~1981~~ 1970 Edition and Addenda through winter ~~1981.~~ 1970. #1

The surveillance requirements for post-RCS opening, modifications and repairs ensure that RCS integrity is demonstrated following conditions that may have affected system integrity.

For normal opening, the integrity of the system, in terms of strength, is unchanged. If the system does not leak at 2335 psig (operating pressure + 100 psi ±100 psi is normal system pressure fluctuation), it will be leak tight during normal operation. E

For repairs or components greater than or equal to 4-inches in ~~2-inch~~ diameter, the thorough nondestructive testing gives a very high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. #

Repairs on components less than 4-inches ~~2-inch~~ in diameter or smaller are relatively minor in comparison and the surface examination assures a similar standard of integrity. In all cases, the leak test will ensure leak tightness during normal operation. #2

3/4.4.11 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the Reactor Coolant System that could inhibit natural circulation core cooling. The OPERABILITY of least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures that the capability exists to perform this function.

The valve redundancy of the Reactor Coolant System vent paths serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. The performances of the specified surveillances will verify the operability of the system.

The function, capabilities, and testing requirements of the Reactor Coolant System vents are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plant Requirements," November 1980.

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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 isolation valves fail to meet single failure criteria, therefore, 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 ECCS components and flowpaths required in Modes 1, 2 and 3 ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming any single active failure consideration. Two SI pumps and one RHR pump operating in conjunction with two accumulators are capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all pipe break sizes up to and including the maximum hypothetical accident of a circumferential rupture of a reactor coolant loop. In addition, the RHR subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period. #4

With the RCS temperature below 350°F, operation with less than full redundant equipment is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ACTION times allow for an orderly sequential shutdown of both units when the inoperability of a component(s) affects both units with equal severity.

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ECCS SUBSYSTEMS (Continued)

The Surveillance Requirements provided for each component ensures that ECCS OPERABILITY is maintained and verified periodically. Surveillance Requirements for throttle valve position stops prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration.

Pump performance requirements are obtained from accident analysis assumptions. Varying flowrates are provided to accommodate testing during modes and alignments. In the case of the 3600 gpm (normal colddown mode) RHR test, differential head is specified in "feet". This criteria will allow for compensation of test data with water density due to varying temperature. cool E

3/4.5.4 REFUELING WATER STORAGE TANK

The OPERABILITY of the refueling water storage tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

The ~~contained~~ indicated water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. E

The temperature limits on the RWST solution ensure that: 1) the solubility of the borated water will be maintained, and 2) the temperature of the RWST solution is consistent with the LOCA analysis. Portable instrumentation may be used to monitor the RWST temperature. #1

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The containment design pressure of 59 psig would not be exceeded if the internal pressure before a major LOCA was as much as 5 psig. The containment is designed to withstand an internal vacuum of 2.5 psig.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the dose guideline values of 10 CFR Part 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 safety 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. #1

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

~~The limitations on containment internal pressure ensure that: (1) the containment structure is prevented from exceeding its design negative pressure differential of 2.5 psig with respect to the outside atmosphere, and (2) the containment peak pressure does not exceed the design pressure of 59 psig during LOCA conditions.~~

~~The maximum peak pressure expected to be obtained from a LOCA event is 55 psig. For the limiting initial positive containment pressure of 3 psig, the total pressure is calculated to be 52.9 psig. This value has been adjusted to 55 psig as the nominal structural design pressure, which is less than design pressure and is consistent with the safety analyses.~~ #1

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CONTAINMENT SYSTEMS

BASES

3/4.6.1.5 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 safety analysis for a LOCA ~~accident.~~ Measurements shall be made at all listed locations, whether by fixed or portable instruments, prior to determining the average air temperature.)E

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 52.9 psig in the event of a LOCA ~~accident.~~)E 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 randomly selected tendons may exclude those which are inaccessible or for which inspection may create a personnel hazard.)#4

The required Special Reports from any engineering evaluation of containment abnormalities shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, the results ~~of the engineering~~ evaluation, and the corrective actions taken.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust ~~isolation valves are required~~ to be closed during a LOCA ~~accident.~~ Maintaining these valves sealed closed during plant operation ensures that excessive quantities of radioactive materials will not be released via the Containment Purge System.)E

The total time the containment purge (vent) system isolation valves may be open during MODES 1, 2, 3, and 4 in a calendar year is a function of anticipated need and operating experience.

Leakage integrity tests with a maximum allowable leakage rate for containment purge supply and exhaust supply valves will provide early indication of resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

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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 capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment and do not reflect the additional redundancy in cooling capability provided by the Containment Cooling System. Pump performance requirements are obtained from the accidents-analysis assumptions.

EMERGENCY

3/4.6.2.2 CONTAINMENT COOLING SYSTEM

)E

The OPERABILITY of the Containment Cooling System ensures that adequate heat removal capacity is available during post-LOCA conditions. The emergency containment coolers are a full capacity system and are redundant to the spray system in terms of heat removal function for design basis accident.

The allowable out-of-service time requirements for the Containment Cooling System have been maintained consistent with that assigned other inoperable ESF equipment and do not reflect the additional redundancy in cooling capability provided by the Containment Spray System.

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

The OPERABILITY of the Emergency Containment Filtering System ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. System components are not subject to rapid deterioration. Visual inspection and operating/performance tests after maintenance, prolonged operation, and at the required frequencies provide assurances of system reliability and will prevent system failure. Filter performance tests are conducted in accordance with the methodology and intent of ANSI N510- 1975.

A Filter Efficiency of 90% is applicable to Turkey Point Safety Analysis. The HEPA filter efficiency combined with greater than 99% elemental iodine removal and 30% organic iodine efficiency for the activated carbon filter will give the applicable overall efficency.

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

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CONTAINMENT ISOLATION VALVES (Continued)

of a release of radioactive material to the containment atmosphere or pressurization of the containment. ~~Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.~~ #2

3/4.6.5 ~~COMBUSTIBLE GAS CONTROL~~ HYDROGEN MONITORS E

The OPERABILITY of the Hydrogen Monitors ensures the detection of hydrogen buildup within containment following a LOCA to allow operator action to reduce the hydrogen concentration below its flammable limit.

3/4.6.6 POST ACCIDENT CONTAINMENT VENT SYSTEM

The OPERABILITY of the Post Accident Containment Vent System ensures the capability for emergency venting of containment following a LOCA to reduce the hydrogen concentration to below its flammable limit.

PACVS systems components are not subject to rapid deterioration, having lifetimes of many years, even under continuous flow conditions. Visual inspection and operating tests provide assurance of system reliability and will ensure early detection of conditions which could cause the system to fail or operate improperly. The performance tests prove that filters have been properly installed, that no deterioration or damage has occurred, and that all components and subsystems operate properly. The tests are performed in accordance with the methodology and intent of ANSI N510-1975 and provide assurance that filter performance has not deteriorated below required specification values due to aging, contamination or other effects.

A filter efficiency of 99% is applicable to Turkey Point Safety Analysis. The HEPA filter efficiency combined with greater than 90% methyliodide removal efficiency for the activated carbon filter will give the applicable overall efficiency.

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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% (1193.5 psig) of its design pressure of 1085 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a Turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section VIII of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 10,670,000 lbs/h which is 111% of the total secondary steam flow of 9,600,000 lbs/h at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-2.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in Secondary Coolant System steam flow and THERMAL POWER required by the reduced Reactor trip settings of the Power Range Neutron Flux channels. The Reactor Trip Setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times (109)$$

Where:

SP = Reduced Reactor Trip Setpoint in percent of RATED THERMAL POWER,

V = Maximum number of inoperable safety valves per steam line,

109 = Power Range Neutron Flux-High Trip Setpoint,

X = Total relieving capacity of all safety valves per steam line in lbs/hour, and

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. Steam can be

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supplied to the pump turbines from either or both units through redundant steam headers. Two D.C. motor operated valves and one A.C. motor operated valve on each unit isolate the three main steam lines from these headers. Both the D.C. and A.C. motor operated valves are powered from safety-related sources. Auxiliary feedwater can be supplied through redundant lines to the safety-related portions of the main feedwater lines to each of the steam generators. Air operated fail closed flow control valves are provided to modulate the flow to each steam generator. Each steam driven auxiliary feedwater pump has sufficient capacity for single and two unit operation 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.

ACTION statement 2 describes the actions to be taken when both auxiliary feedwater trains are inoperable. The requirement to verify the availability of both standby feedwater pumps is to be accomplished by verifying that both pumps have successfully passed their monthly surveillance tests within the last surveillance interval. The requirement to complete this action before beginning a unit shutdown is to ensure that an alternate feedwater train is available before putting the affected unit through a transient. If no alternate feedwater trains are available, the affected unit is to stay at the same condition until an auxiliary feedwater train is returned to service, and then invoke ACTION statement 1 for the other train. If both standby feedwater pumps are made available before one auxiliary feedwater train is returned to an OPERABLE status, then the affected unit(s) shall be placed in at least HOT STANDBY within 6 hours* and HOT SHUTDOWN within the following 6 hours. E #1

ACTION statement 3 describes the actions to be taken when a single auxiliary feedwater pump is inoperable. The requirement to verify that two independent auxiliary feedwater trains are OPERABLE is to be accomplished by verifying that the requirements for Table 3.7-3 have been successfully met for each train within the last surveillance interval. ~~ACTION statement 4 states that~~ The provisions of Specification 3.0.4 are not applicable to the third auxiliary feedwater pump provided it has not been inoperable for longer than 30 days. This means that a unit(s) can change OPERATIONAL MODES during a unit(s) heatup with a single auxiliary feedwater pump inoperable as long as the requirements of ACTION statement 3 are satisfied. E

The monthly testing of the auxiliary feedwater pumps will verify their operability. Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps.

3/4.7.1.3 CONDENSATE STORAGE TANK

There are two (2) seismically designed 250,000 gallons condensate storage tanks. A minimum of 185,000 gallons is maintained in each tank. The OPERABILITY of the condensate storage tank with the minimum water volume ensures that

*If this ACTION applies to both units simultaneously be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. #1

CONDENSATE STORAGE TANK (Continued)

sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for approximately 23 hours or maintain the Reactor Coolant System at HOT STANDBY conditions for 15 hours and then cool down the Reactor Coolant System to below 350°F at which point the Residual Heat Removal System may be placed in operation.

3/4.7.1.4 SPECIFIC ACTIVITY

The limit on secondary coolant specific activity is based on a postulated release of secondary coolant equivalent to the contents of three steam generators to the atmosphere due to a net load rejection. The limiting dose for this case would result from radioactive iodine in the secondary coolant. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary making allowance for plate-out and retention in water droplets. The inhalation thyroid dose at the site boundary is then;

$$\text{Dose (Rem)} = C \circ V \circ B \circ DFC \circ X/Q \circ 0.1$$

Where:

	C	=	secondary coolant dose equivalent I-131 specific activity
		=	0.2 curies/m^3 (μCi/cc) or 0.1 Ci/m^3, each unit
	V	=	equivalent secondary coolant volume released = 214 m^3
	B	=	breathing rate = 3.47×10^{-4} m^3/sec.
	X/Q	=	atmospheric dispersion parameter = 1.54×10^{-4} sec/m^3
	0.1	=	equivalent fraction of activity released
	DCF	=	dose conversion factor, Rem/Ci

The resultant thyroid dose is less than 1.5 Rem.

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 blow down 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 safety analyses.

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The term operable/operability has been modified so as to positively avoid any implication or connotation that the Standby Feedwater System is designed or documented to meet safety system requirements such as seismic loads, environmental qualification, safety grade emergency power supply, etc.

3/4.7.1.6 STANDBY FEEDWATER SYSTEM

The purpose of this specification and the supporting surveillance requirements is to provide for administrative controls which will assure ~~availability~~ operability and performance of the non-safety grade Standby Feedwater System. The Standby Feedwater System consists of commercial grade components designed and constructed to industry and FPL standards of this class of equipment located in the outdoor plant environment typical of FPL facilities system wide. The system is expected to perform with high reliability, i.e., comparable to that typically achieved with this class of equipment. FPL intends to maintain the system in good operating condition with regard to appearance, structures, supports, component maintenance, calibrations, etc. #1

The function of the Standby Feedwater System for OPERABILITY determinations is that it can be used as a backup to the Auxiliary Feedwater (AFW) System in the event the AFW System does not function properly. The system would be manually started, aligned and controlled by the operator when needed. In the event of a loss of offsite power the pumps can be powered via the non-safety grade diesel generators connected to the ~~non-vital~~ non-safety related 4160 volt bus. #1 E

A supply of 60,000 gallons from the Demineralized Water Storage Tank for the Standby Feedwater Pumps is sufficient water to remove decay heat from the reactor for six (6) hours for a single unit or two (2) hours for two units. This was the basis used for requiring 60,000 gallons of water in the non-safety grade Demineralized Water Storage Tank and is judged to provide sufficient time for restoring the AFW System or establishing make-up to the Demineralized Water Storage Tank.

The motor driven Standby Feedwater Pumps are not designed to NRC requirements applicable to Auxiliary Feedwater Systems and not required to satisfy design basis events requirements. These pumps may be out of service for up to 24 hours before initiating formal notification because of the extremely low probability of a demand for their operation.

The guidelines for NRC notification in case of both pumps being out of service for longer than 24 hours are provided in applicable plant procedures, as a ~~VOLUNTARY~~ courtesy 4-hour notification. E

Adequate demineralized water for the standby feedwater system will be verified once per 24 hours. The Demineralized Water Storage Tank provides a source of water to several systems and therefore, requires daily verification.

The standby feedwater pumps will be verified ~~available~~ OPERABLE monthly on a STAGGERED TEST BASIS by starting and operating them in the recirculation mode typically from their normal power supply. Also, during each unit's refueling outage, the respective standby feedwater pump will be powered from the unit's C bus utilizing Units 1 and 2 non-safety grade diesel generators and flow E

STANDBY FEEDWATER SYSTEM (Continued)

tested to the nuclear unit's steam generators. Prior to this test, the refueling unit's C bus will be de-energized and the necessary loads will be transferred to the other unit's C bus.

This surveillance regimen will thus demonstrate operability. ~~availability~~ of the entire flow path, backup non-safety grade power supply and pump associated with a unit at least each refueling outage. The pump, motor driver, and normal power supply availability would typically be demonstrated by operation of the pumps in the recirculation mode monthly on a staggered test basis. E E

use FULL CAPITALS

3/4.7.2 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 provided by the required equipment ~~of this system~~, assuming a single active failure, is consistent with the assumptions used in the safety analyses. ~~One pump and two heat exchangers provide the heat removal capability for accidents that have been analyzed.~~ #7

ADD ATTACHED INSERT

3/4.7.3 INTAKE COOLING WATER SYSTEM

The OPERABILITY of the Intake Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The design and operation ~~cooling capacity~~ of this system, assuming a single active failure, ensures cooling capacity ~~is~~ consistent with the assumptions used in the safety analyses. #5

~~3/4.7.4 ULTIMATE HEAT SINK~~

~~The limitations on the ultimate heat sink temperature ensure that sufficient cooling capacity is available either: (1) to provide normal cooldown of the facility or (2) to mitigate the effects of accident conditions within acceptable limits.~~

~~The limitations on minimum water level and maximum temperature are based on providing a 30-day cooling water supply to safety-related equipment without exceeding its design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.~~ #1

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EMERGENCY

3/4.7.5 CONTROL ROOM VENTILATION SYSTEM

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Emergency

The OPERABILITY of the Control Room Ventilation System ensures that: (1) the ambient air temperature does not exceed the allowable temperature for continuous-duty rating for the equipment and instrumentation cooled by this system, and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

System components are not subject to rapid deterioration, having lifetimes of many years, even under continuous flow conditions. Visual inspection and operating tests provide assurance of system reliability and will ensure early detection of conditions which could cause the system to fail or operate improperly. The filters performance tests prove that filters have been properly installed, that no deterioration or damage has occurred, and that all components and subsystems operate properly. The tests are performed in accordance with the methodology and intent of ANSI N510 (1975) and provide assurance that filter performance has not deteriorated below returned specification values due to aging, contamination, or other effects.

3/4.7.6 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.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to each safety-related system during an earthquake or severe transient. Therefore, the required inspection interval varies inversely with the observed snubber failures 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 visual inspection is clearly established and remedied for the snubber and for any other snubbers that may be generically susceptible, and verified operable by inservice functional testing, that snubber may be exempted from being counted as inoperable for the purposes of establishing the next visual inspection interval. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

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INSERT TO BASES FOR 3/4.7.2 ON CCW

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The plant design includes three CCW heat exchangers. However, analysis results have shown that one pump and the combined performance of two heat exchangers will meet the cooling requirements assumed in the accident analysis. A surveillance program monitors the system capability to meet these requirements. The 31 day surveillance establishes the heat exchanger performance characteristics. The 12 hour surveillance monitors intake cooling water inlet temperature and correlates it with the heat exchanger performance characteristics and other system parameters to verify heat removal capability. To provide additional assurance of heat exchanger availability, all three heat exchangers will be included in this monitoring program. #7

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SNUBBERS (Continued)

When a snubber is found inoperable, an evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any Safety Related System or component has been adversely affected by the inoperability of the snubber. The 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 refueling SHUTDOWNS. Observed failure of these sample snubbers shall require functional testing of additional units.

In cases where the cause of the functional failure has been identified additional testing shall be based on manufacturer's or engineering recommendations. As applicable, this additional testing increases the probability of locating possible inoperable snubbers without testing 100% of the safety-related snubbers.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replaced, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.7 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(a)(3) 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.

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PLANT SYSTEMS

BASES

3/4.7.8 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.

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.

3/4.7.9 FIRE RATED ASSEMBLIES

The functional integrity of the fire rated assemblies and barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacer portions of the facility. These design features minimize the possibility of a sing fire rapidly involving several areas of the facility prior to detection and extinguishing of the fire. The fire barrier penetrations are a passive element in the facility Fire Protection Program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. ~~For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.~~ #1

During periods of time when a barrier is not functional, either: (1) a continuous fire watch is required to be maintained in the vicinity of the affected barrie or (2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established until the barrier is restored to functional status.

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3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BOROn CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. ~~The locking closed of~~ the required valves closed during refueling operations ~~precludes~~ the possibility of uncontrolled boron dilution of the filled portion of the RCS is precluded. This action prevents flow to the RCS of unborated water by closing flow paths from sources of unborated water. #3

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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. There are four source range neutron flux channels, two primary and two backup. All four channels have visual and alarm indication in the control room and interface with the containment evacuation alarm system. The primary source range neutron flux channels can also generate reactor trip signals and provide audible indication of the count rate in the control room and containment. At least one primary source range neutron flux channel to provide the required audible indication, in addition to its other functions, and one of the three remaining source range channels shall be OPERABLE to satisfy the LCO.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment 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.

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3/4.9.6 MANIPULATOR CRANE

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 reactor vessel are protected from excessive lifting force in the event they ~~are inad~~vertently engaged during lifting operations.

The requirement that the auxiliary hoist load indicator be used to prevent lifting excessive loads will require a manual action. The auxiliary hoist load indicator does not include any automatic ~~mechanical~~ or electrical interlocks that prevent lifting loads in excess of 600 pounds.

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3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE 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 racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety 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 vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the core to minimize the effect of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT VENTILATION ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment ventilation 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.

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3/4.9.10 and 3/4.9.11 WATER LEVEL - REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 HANDLING OF SPENT FUEL CASK

Limiting spent fuel decay time to a minimum of 1,525 hours prior to moving a spent fuel cask into the spent fuel pit will ensure that potential offsite doses are a fraction of 10 CFR Part 100 limits should a dropped cask strike the stored fuel assemblies.

The restriction to allow only a single element cask to be moved into the spent fuel pit will ensure the maintenance of water inventory in the unlikely event of an uncontrolled cask descent. Use of a single element cask which nominally weighs about twenty-five tons will also increase crane safety margins by about a factor of four. |E

Requiring that spent fuel decay time be at least 120 days prior to moving a fuel assembly outside the fuel storage pit in a shipping cask will ensure that potential offsite doses are a fraction of 10 CFR 100 limits should a dropped cask and ruptured fuel assembly release activity directly to the atmosphere.

3/4.9.13 RADIATION MONITORING

The OPERABILITY of ~~the~~ a containment radiation monitor~~s~~ ensures continuous monitoring of radiation levels to provide immediate indication of an unsafe condition.)#1

3/4.9.14 SPENT FUEL STORAGE

The spent fuel storage racks provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison to assure k_{eff} is equal to or less than ~~0.90~~ 0.95 for normal operations and postulated accidents.)#1

The spent fuel racks are divided into two regions. Region I racks have a 10.6 inch center-to-center spacing and Region II racks have a 9.0 inch center-to-center spacing. Because of the larger center-to-center spacing and poison (B^{10}) concentration of Region I cells, the only restriction for placement of fuel is that the initial fuel assembly enrichment is equal to or less than 4.5 weight percent of U-235. The limiting value of U-235 enrichment is based upon the assumptions in the spent fuel safety analyses and assures that the limiting criteria for criticality is not exceeded. Prior to placement

SPENT FUEL STORAGE (Continued)

in Region II cell locations, strict controls are employed to evaluate burnup of the spent fuel assembly. Upon determination that the fuel assembly meets the burnup requirements of Table 3.9-1, placement in a Region II cell is authorized. These positive controls assure the fuel enrichment limits assumed in the safety analyses will not be exceeded.*

*This Technical Specification is applicable upon installation of the new two-region high density spent fuel racks.

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 occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to measure control rod worth.

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}~~ average coolant temperature slightly lower than normally allowed so that the fundamental nuclear characteristics of the core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS ~~T_{avg}~~ average coolant temperature to fall slightly below the minimum temperature of Specification 3.1.1.4. E

3/4.10.4 (This specification number is not used.)

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 are 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, cannot be observed if the Position Indication Systems remain OPERABLE.

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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 to UNRESTRICTED AREAS 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 in UNRESTRICTED AREAS will result in exposures within: (1) the objectives of Appendix I, 10 CFR Part 50, to a MEMBER OF THE PUBLIC, and (2) the limits of 10 CFR Part 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.

This specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4077 (September 1984), in HASL Procedures Manual, HASL-300 (revised annually) and in Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive material in liquid effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The dose calculation methodology and parameters in the ODCM implement the requirements in 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 a MEMBER OF THE PUBLIC 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," March, 1976 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.

DOSE (Continued)

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Systems, the liquid effluents from the shared system are to be proportional among the units sharing that system.

3/4.11.1.3 LIQUID RADWASTE TREATMENT SYSTEM

The OPERABILITY of the Liquid Radwaste Treatment System ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the 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 Liquid Radwaste Treatment System were specified as a suitable fraction of the dose ~~design~~ objectives set forth in Appendix I, 10 CFR Part 50 for liquid effluents. e 1E

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. For units with shared Radwaste Systems the liquid effluents from the shared system are to be proportioned among the units sharing that system.

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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 and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC 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 MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. Examples of calculations for such MEMBERS OF THE PUBLIC, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the whole body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive material in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4077 (September 1984), in HASL Procedures Manual, HASL-300 ~~(revised annually)~~ and in Hartwell, J.K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975). E

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established

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DOSE-NOBLE GASES (Continued)

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," March 1976, 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 and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions.

This specification applies to the release of radioactive materials in gaseous effluents from each unit 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.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM

This specification is provided to implement the requirements of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in 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 a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters 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," March 1967, 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 Iodine-131 Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of the 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.

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DOSE - IODINE-133, TRITIUM, AND RADIOACTIVE MATERIAL IN PARTICULATE FORM (Continued)

This specification applies to the release of radioactive materials in gaseous effluents from each unit 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.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT SYSTEM

The OPERABILITY of the GAS DECAY TANK 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 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the 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 objectives set forth in Appendix I, 10 CFR Part 50, for gaseous effluents.

This specification applies to the release of radioactive materials in gaseous effluents from each unit 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.

RADIOACTIVE EFFLUENTS

BASES

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 GAS DECAY TANK SYSTEM (as measured in the inservice gas decay tank) is maintained below the flammability limits of hydrogen and oxygen. 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 DECAY TANKS

The tanks included in this specification are those tanks for which the quantity of radioactivity contained is not limited directly or indirectly by another Technical Specification. Restricting the quantity of radioactivity contained in each Gas Decay Tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting whole body exposure to a MEMBER OF THE PUBLIC at the nearest SITE BOUNDARY will not exceed 0.5 rem.

3/4.11.3 SOLID RADIOACTIVE WASTES

This specification implements the requirements of 10 CFR 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, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 10 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources exceed 25 mrems to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the units ~~(including outside~~

TOTAL DOSE (Continued)

~~storage tanks, etc.)~~ are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 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 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC·staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

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3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The Radiological Environmental Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure of MEMBERS OF THE PUBLIC resulting from the plant operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and 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 the modeling of the environmental exposure pathways.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum 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 an a posteriori (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in Currie, L. A., "Lower Limit of Detection: Definition and Elaboration of a Proposed Position for Radiological Effluent and Environmental Measurements," NUREG/CR-4007 (September 1984), in HASL Procedures Manual, HASL-300 ~~(revised annually)~~ and Hartwell, J. K. "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975). E

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the Radiological Environmental Monitoring Program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from 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 50 m^2 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 made: (1) 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/m^2.

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3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50. This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

The LCO is modified by a footnote which accounts for the process of providing samples for multiple testing programs to a single laboratory without requiring reporting of issues which do not pertain to the Turkey Point Radiological Environmental Monitoring Program. #1

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SECTION 5.0

DESIGN FEATURES

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T. S. 3/4.3.2

Justifications:

1. The CTS does not address "allowable" values. The 1986 version of RTS adopted the STS 2-column format containing nominal setpoints and allowables. The RTS allowables were best estimate numbers determined by Westinghouse using generic tolerances. However, FPL would require additional analysis to adequately support these values.

2. This change clarifies an oversight. It has been revised to be consistent with Functional Unit 4.d of Table 3.3-2.

3. Use of jumpers; fuses to place channel in trip condition was agreed to by NRC technical reviewer in FPL/NRC working meetings.

4. For some channels, (e.g., containment press hi), an analog channel operational test is not applicable.

5. There is no testing of relays that can be accomplished beyond that done by ACTUATION LOGIC TEST.

6. A monthly TRIP ACTUATING DEVICE OPERATIONAL TEST cannot be performed without initiating an actual high pressure condition. This is performed during refueling calibration.

7. With deletion of the allowable value column, the trip setpoints are revised to reflect the CTS values.

8. The automatic actuation logic and actuation relays are required surveillances in MODES 3 and 4; however, they are not testable until MODE 3. This problem exists with items 1.b, 1.d through 1.f, 4.b, 4.d, 8.a and 8.b.

9. Several instruments do not have independent channels measuring the same parameter. To perform a CHANNEL CHECK on these instruments, a known source is needed to manually check each instrument with an independent channel.

10. Functional Unit 9.a is not testable for ACTUATION LOGIC TEST of radiation monitors.

11. High Steam Flow-Coincident with Steam Generator Pressure-Low or T_{avg} - Low can only be blocked for safety injection. Steam Line Isolation cannot be blocked.

12. With deletion of the allowable value column, the trip setpoints are revised to reflect the CTS values.

T. S. 3/4.3.2 (continued)

13. BASES - The deleted wording provides no necessary information.

14. There are no control room isolation functions associated with high-containment radioactivity monitors during fuel movement in the spent fuel pool.

T. S. <u>3/4.3.3.1</u>

Justifications:

1. The monitor numbers were removed to prevent making a License Amendment if instrument make/model changes are made in the future.

2. Several instruments do not have independent channels measuring the same parameter. To perform a CHANNEL CHECK on these instruments, a known source is needed to manually check each instrument with an independent channel.

T. S. 3/4.3.3.3

Justifications:

1. Two different specifications exist for the same requirement 3.3.3.3 and 3.6.5. Specification 3.0.4 is not applicable to Table 3.3-5, but it is applicable to Specification 3.6.5. This is a source of confusion that needs to be eliminated.

2. The accident monitoring instruments do not provide trip/active functions; therefore, there is no need to conduct the ANALOG CHANNEL OPERATIONAL TEST which requires simulated input signals to test interlock/alarm/trip functions.

3. The refueling surveillance is consistent with the STS.

4. The note describes the only possible check that can be performed because there is no visible indication during normal conditions.

5. This note allows the neutron detector to be excluded from channel calibration. This calibration is not practical.

6. Comparing wide range instruments during normal operation is not a meaningful CHANNEL CHECK. A source must be used to obtain a response. This change is consistent with STS.

7. This surveillance is inconsistent with the LCO's which require these channels only in Modes 1, 2 and 3.

8. The monitor numbers were removed to prevent making a License Amendment if instrument make/model changes are made in the future.

9. For consistency of reporting requirements, 30 days is allowed for submittal of the report.

T. S. 3/4.3.3.4

Justifications:

1. Failure to comply with the ACTION Statement is reportable via 10 CFR 50.73. This was a CTS requirement, not an STS requirement.

2. Containment entry to inspect the fire zone is consistent with the containment temperature surveillance frequency (4.6.1.5) and is justified based on ALARA considerations and the likelihood of recognizing fire concerns through other instrumentation/equipment feedback.

3. Fire watch patrol has been replaced by additional instrumentation as part of Appendix R upgrades.

T. S. 3/4.3.3.5

Justifications:

1. The monitor numbers were removed to prevent making a License Amendment if instrument make/model changes are made in the future.

2. Several instruments do not have independent channels measuring the same parameter. To perform a CHANNEL CHECK on these instruments, a known source is needed to manually check each instrument with an independent channel.

3. Applicability should be limited to those situations when releases are made via these pathways. This is consistent with CTS (Amendment 103/97).

T. S. 3/4.3.3.6

Justifications:

1. The iodine and particulate samplers are counted filters, not online monitors. Their function is covered in Table 4.11-2.

2. Several instruments do not have independent channels measuring the same parameters. To perform a CHANNEL CHECK on these instruments, a known source is needed to manually check each instrument with an independent channel.

3. This is a continuous release pathway and should be monitored daily as well as prior to any release.

4. Four (4) hours would require upgrades to equipment. Eight (8) hours is a reasonable amount of time to perform this test.

5. The monitor numbers were removed to prevent making a License Amendment if instrument make/model changes are made in the future.

T.S. <u>3/4.4.1</u>

Justifications:

1. BASES - This clarification is consistent with the CTS.

T. S. 3/4.4.1.2

Justifications:

1. Refer to attached Westinghouse letters.

2. The single active failure wording is not applicable to Turkey Point due to electrical system design.

3/4.4.1.2

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Fuel Division

Box 3912
Pittsburgh Pennsylvania 15230-3912

84FP*-G-126
October 8, 1984
Keywords: FPL Rod-Withdrawal
Reference: 1) FPL-84-729, 8/16/84
2) NS-EPR-2935, 7/9/84
3) FP-FP-697
82FP*-G-011

Mr. Steve Craig
Florida Power and Light Company
P.O. Box 013100
Miami, FL 33152

Dear Mr. Craig:

FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4
<u>PREVENTION OF ROD WITHDRAWAL FROM SUBCRITICAL DURING MODE 3 OPERATION</u>

The purpose of this memo is to provide additional requested information regarding an inconsistency between the safety analysis and the Tech Specs with respect to the number of operating reactor coolant pumps when in Mode 3 (or the equivalent) as defined in the Tech Specs. Reference 1 provided the latest update on this issue. The purpose of this letter is to provide confirmation of the safety analysis assumptions and FPL's proposed administrative procedure to address this issue, as discussed with Mr. Rick Mende, Reactor Engineer at Turkey Point.

As discussed in Reference 2, a letter from Westinghouse to the NRC, the only event impacted by this issue is the RCCA bank withdrawal from subcritical. This accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs resulting in a power excursion. Such a transient could be caused by a malfunction of the automatic rod control system or by operator error. The analysis assumes a reactivity insertion rate of 75 pcm/sec, greater than the maximum resulting from the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

The most recent RCCA bank withdrawal from subcritical analysis for Turkey Point Units 3 and 4 was performed as part of the positive MTC study in 1981 (Reference 3), and assumes that all three reactor coolant pumps are operating. However, the Turkey Point Tech Specs state that "In Hot Shutdown at least two Reactor Coolant Loops shall be operable and at least one Reactor Coolant Loop shall be in operation." (T.S. 3.4.1.d, pg 3.4-2A). Thus, the Tech Specs are inconsistent with the analysis assumption. As noted in Reference 2, it is not a realistic requirement to have all three reactor coolant pumps operating when the plant is cooling down prior to going to Cold Shutdown. In addition, administrative procedures are preferable to physical prevention of rod withdrawal for Turkey Point, since the plant

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The Exclusion Area shall be as shown in Figure 5.1-1 (E)

LOW POPULATION ZONE

5.1.2 The Low Population Zone shall be as shown in Figure 5.1-1.

MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS

5.1.3 Information regarding radioactive gaseous and liquid effluents, which will allow identification of structures and release points as well as definition of UNRESTRICTED AREAS within the SITE BOUNDARY that are accessible to MEMBERS OF THE PUBLIC, shall be as shown in Figure 5.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The 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 = 116 feet.

b. Nominal inside height = 170.6 feet.

c. Minimum thickness of concrete walls = 3.75 feet.

d. Minimum thickness of concrete roof = 3.25 feet.

e. Minimum thickness of concrete floor pad = 10.5 feet.

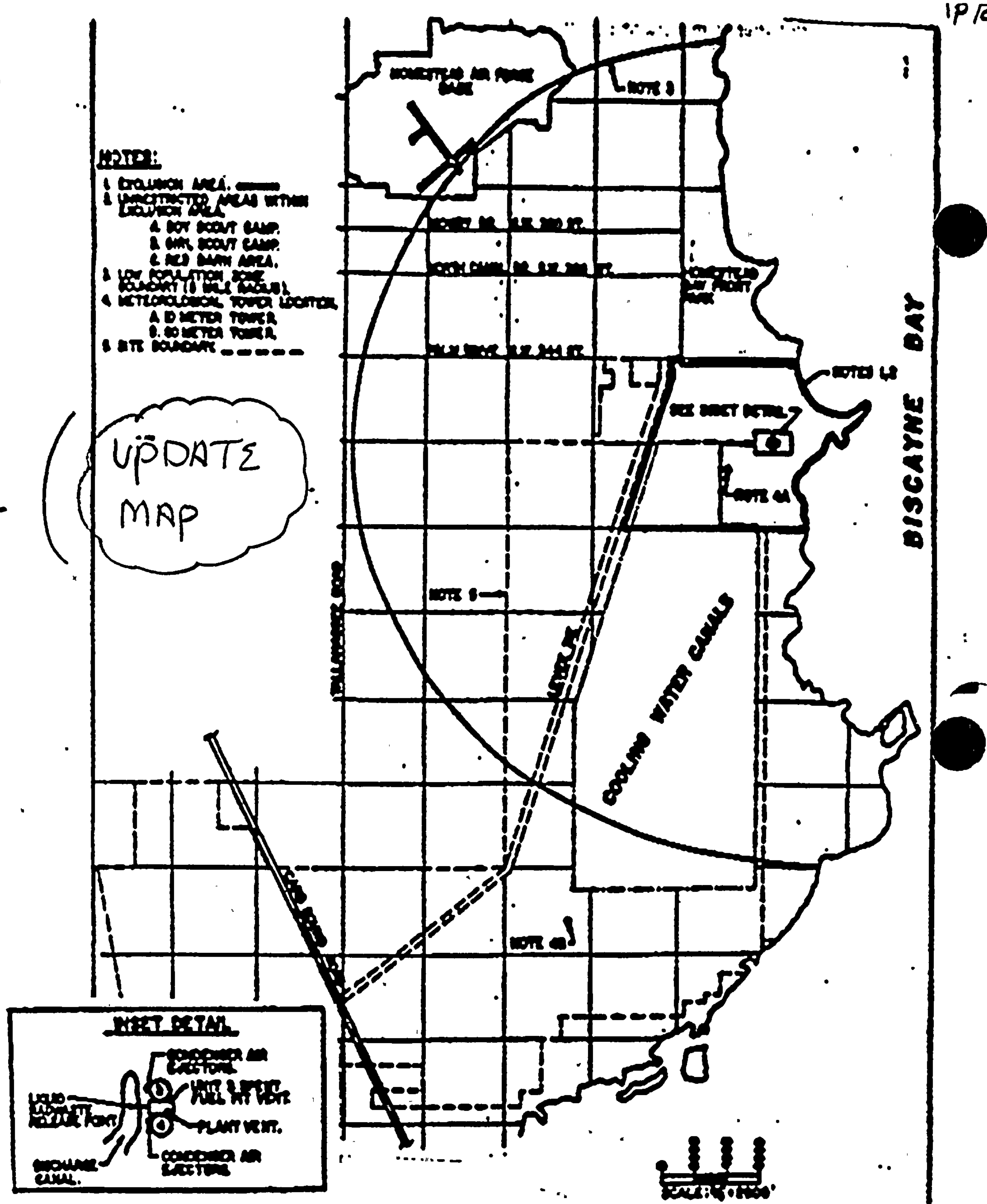
f. Nominal thickness of steel liner = 0.25 inches.

g. Net free volume = 1,550,000 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The containment building is designed and shall be maintained for a maximum internal pressure of 59 psig and a temperature of 283°F. The containment building is also structurally designed to withstand an internal vacuum of 2.5 psig.

1PR



#1 UPDATE MAP

FIGURE 5.1-1 SITE AREA MAP

, except that replacement of fuel rods by filler rods consisting of stainless steel, or by vacant rod positions, may be made in fuel assemblies if justified by cycle-specific reload analysis using NRC-approved methodology.

DESIGN FEATURES

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies with each fuel assembly nominally containing 204 fuel rods clad with Zircaloy-4. The reactor core contains approximately 71 metric tons of uranium in the form of natural or slightly enriched uranium dioxide pellets. Each fuel rod shall have a nominal active fuel length of 144 inches. ~~The initial core loading shall have a maximum enrichment of 3.1 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.1 weight percent U-235.~~ #2 #1 #2

CONTROL ROD ASSEMBLIES

5.3.2 The core shall contain 45 full-length control rod assemblies. The full-length control rod assemblies shall contain a nominal 142 inches of absorber material. The absorber material shall be silver, indium, and cadmium. All control rods shall be ~~All control rods shall be~~ clad with stainless steel tubing. E

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 4.1 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 nominal water and steam volume of the Reactor Coolant System is 9343 cubic feet at a nominal T_{avg} of 574.2°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological towers shall be located as shown on Figure 5.1-1.

Should more than 30 individual fuel rods in the core, or 10 fuel rods in any one assembly, be replaced per refueling, a Special Report discussing the rod replacements will be submitted to the Commission within 30 days after cycle startup.

PR

5.6 FUEL STORAGE

5.6.1 CRITICALITY

5.6.1.1 The spent fuel storage racks are designed (e) to provide safe subcritical storage of fuel assemblies by providing sufficient center-to-center spacing or a combination of spacing and poison and shall be maintained with: (E)

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of 2.55% $\Delta k/k$ for uncertainties for single region spent fuel storage racks.

b. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance in region 1 of 2.66% $\Delta k/k$ and in region 2 of 2.94% $\Delta k/k$ for uncertainties for two region fuel storage racks.

c. A nominal 13.7 inch center-to-center distance between fuel assemblies placed in the single-region storage racks. A nominal 10.6 inch center-to center distance for Region 1 and 9.0 inch center-to-center distance for Region 2 for two region fuel storage racks.

d. Fuel assemblies stored in the single-region spent fuel storage racks shall contain no more than 4.1 weight percent of U-235.

e. After installation of the two-region high density spent fuel storage racks, the maximum enrichment loading for fuel assemblies is 4.5 weight percent of U-235.

5.6.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.

5.6.1.3 Credit for burnup is taken in determining placement locations for spent fuel in the two-region spent fuel racks.* Administrative controls are employed to evaluate the burnup of each spent fuel assembly stored in areas where credit for burnup is taken. The burnup of spent fuel is ascertained by careful analysis of burnup history, prior to placement into the storage locations. Procedures shall require an independent check of the analysis of suitability for storage. A complete record of such analysis is kept for the time period that the spent fuel assembly remains in storage onsite.

DRAINAGE

5.6.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 621** fuel assemblies in one region storage racks or 1404 in two region storage racks

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.

*During rack installation, it will be necessary to temporarily store Region I fuel in the Region II spent fuel racks. Administrative controls will be utilized to maintain a checkerboard storage configuration, i.e., alternate cell occupation, in the Region II racks.

**The fuel assembly storage capacity for Unit 4 single region storage racks is 614.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT	
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.	
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.	
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.	
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.	
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.	
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.	
	150 ~~40~~ leak tests.	Pressurized to ≥ 2435 psig.	#1
	5 hydrostatic pressure tests.	Pressurized to ≥ 3100 psig.	
Secondary Coolant System	6 loss of secondary pressure	Loss of Secondary pressure	
	50 leak tests	Pressurized to ≥ 1085 psig	#1
	35 ~~10~~ hydrostatic pressure tests.	Pressurized to ≥ 1356 psig.	

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6.1 RESPONSIBILITY

6.1.1 The Plant Manager - Nuclear shall be responsible for overall unit(s) operation and shall delegate in writing the succession to this responsibility during his absence. (E)

6.1.2 The Plant Supervisor - Nuclear (or during his absence from the control room, a designated individual) shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President ~~- Nuclear Energy~~ shall be reissued to all station personnel on an annual basis. (E)

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for facility operation and corporate management. The onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to an including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Topical Quality Assurance Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The Senior Vice President-Nuclear shall be responsible for overall plant nuclear safety, and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety.

c. The Plant Manager-Nuclear shall be responsible for overall unit safe operation and shall have control over those onsite activities necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the health physics manager shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

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FACILITY
~~UNIT~~ STAFF

6.2.2 The facility ~~unit~~ organization shall be subject to the following:

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 Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips. In addition, while the unit is in MODE 1, 2, 3, or 4, at least one licensed Senior Operator shall be in the control room;

d. A Health Physics Technician* shall be on site when fuel is in the reactor;

e. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation;

f. A site Fire Brigade of at least five members* shall be maintained on site at all times. The Fire Brigade shall not include the Shift Supervisor and the two 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; and

g. Administrative procedures shall be developed and implemented to limit the working hours of unit(s) staff who perform safety-related functions (e.g., licensed Senior Operators, licensed Operators, health physicists, auxiliary operators, and key maintenance personnel). (E)

or

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the unit is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance, or major plant modification, on a temporary basis the following guidelines shall be followed:

*The Health Physics Technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

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UNIT STAFF (Continued)

1. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

2. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

3. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

4. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager - Nuclear or his ~~deputy~~ designee, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager - Nuclear or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized. E

h. The Operations Supervisor shall hold a Senior Reactor Operator License.

i. The Operations Superintendent shall ~~hold a Senior Reactor Operator License.~~ #1

either hold or have held a Senior Reactor Operator License on the Turkey Point Plant, or have held a Senior Reactor Operator License on a similar plant (ie, another pressurized water reactor).

✓PR

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION		
	BOTH UNITS IN MODE 1, 2, 3, or 4	BOTH UNITS IN MODE 5 or 6 OR DEFUELED	ONE UNIT IN MODE 1, 2, 3, or 4 AND ONE UNIT IN MODE 5 or 6 or DEFUELED
PSN	1	1	1
SRO	1	none**	1
RO	3*	2*	3*
AO	3*	3*	3*
STA	1***	none	1***

PSN - Plant Supervisor Nuclear with a Senior Operator license
SRO - Individual with a Senior Operator license
RO - Individual with an Operator license
AO - Auxiliary Operator
STA - Shift Technical Advisor

The shift crew composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Plant Supervisor Nuclear from the control room while a unit is in MODE 1, 2, 3, or 4, an individual (other than the Shift Technical Advisor) with a valid Senior Operator license shall be designated to assume the control room command function. During any absence of the Plant Supervisor Nuclear from the control room while both units are in MODE 5 or 6, an individual with a valid Senior Operator license or Operator license shall be designated to assume the control room command function.

*At least one of the required individuals must be assigned to the designated position for each unit.

**At least one licensed Senior Operator or licensed Senior Operator Limited to Fuel Handling must be present during CORE ALTERATIONS on either unit, who has no other concurrent responsibilities.

***The STA position shall be manned in MODES 1, 2, 3, and 4 unless the Plant Supervisor Nuclear or the individual with a Senior Operator license meets the qualifications for the STA as required by the NRC.

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6.2.3 SHIFT TECHNICAL ADVISOR

6.2.3.1 The Shift Technical Advisor shall provide advisory technical support to the Plant Supervisor Nuclear in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit. The Shift Technical Advisor shall have a bachelor's degree or equivalent in a scientific or engineering discipline and shall have received specific training in the response and analysis of the unit for transients and accidents, and in unit design and layout, including the capabilities of instrumentation and controls in the control room. S the s S (E)

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and the Operations Superintendent whose requirement for a Senior Reactor Operator License is as stated in Specification 6.2.2.i. The licensed Operators and Senior Operators shall also meet or exceed the minimum qualifications of the supplemental requirements specified in ~~Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees.~~ 10 CFR 55 and ANSI 3.1,1981. #1 #2

6.3.2 When the Health Physics Supervisor does not meet the above requirements, compensatory action shall be taken which the Plant Nuclear Safety Committee determines and the NRC office of Nuclear Reactor Regulation concurs that the action meets the intent of Specification 6.3.1.

6.4 TRAINING

6.4.1 A retraining ~~and replacement~~ training program for the facility staff shall be maintained under the direction of the Training Superintendent and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and ~~Appendix A of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees~~ 10 CFR 55 and ANSI 3.1, 1981, and shall include familiarization with relevant industry operational experience. #1

6.4.2 A training program for the fire brigade shall be maintained under the direction of the Fire Protection ~~Administrator~~ Supervisor and shall meet or exceed the requirements of 10 CFR 50.48 and 10 CFR 50 Appendix R. (E)

6.5 REVIEW AND AUDIT

6.5.1 PLANT NUCLEAR SAFETY COMMITTEE (PNSC)

FUNCTION

6.5.1.1 The PNSC shall function to advise the Plant Manager - Nuclear on all matters related to nuclear safety.

PR

Member: Plant Manager - Nuclear
Member: Operations Superintendent
Member: Operations Supervisor
Member: Maintenance Superintendent
Member: Instrument & Control Supervisor
Member: Reactor Supervisor
Member: Health Physics Supervisor
Member: Technical Supervisor
Member: Chemistry Supervisor
Member: Quality Control Supervisor
Member: Assistant Plant Supt. Electrical
Member: Assistant Plant Supt. Mechanical

COMPOSITION

6.5.1.2 The PNSC shall be composed of the:

~~Chairman: Plant Manager - Nuclear~~
~~Vice Chairman: Operations Superintendent - Nuclear~~
~~Member: Technical Department Supervisor~~
~~Member: Maintenance Superintendent - Nuclear~~
~~Member: Instrument and Control Supervisor~~
~~Member: Reactor Supervisor~~
~~Member: Health Physics Supervisor~~

The chairman shall be a member of the PNSC and shall be designated in writing. #1

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PNSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as members in PNSC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PNSC shall meet at least once per calendar month and as convened by the PNSC Chairman. ~~or Vice Chairman.~~ #1

QUORUM

6.5.1.5 The quorum of the PNSC necessary for the performance of the PNSC responsibility and authority provisions of these Technical Specifications shall consist of the Chairman ~~or Vice Chairman~~ and four members including alternates. #1

RESPONSIBILITIES

6.5.1.6 The PNSC shall be responsible for:

a. Review of: (1) all proposed administrative and emergency operating procedures ~~required by Specification 6.8.1.a-g~~ and changes thereto, ~~(2) all proposed programs required by Specification 6.8.4 and changes thereto,~~ and ~~(3)~~ (2) any other proposed procedures or changes thereto as determined by the Plant Manager - Nuclear; ~~to affect nuclear safety;~~ #1

b. Review of all proposed tests and experiments that affect nuclear safety;

c. Review of all proposed changes to Appendix "A" Technical Specifications;

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety;

RESPONSIBILITIES (Continued)

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 Senior Vice President-Nuclear ~~Group Vice President of Nuclear Energy~~ and to the Chairman of the Company Nuclear Review Board; E

f. Review of all REPORTABLE EVENTS;

g. Review of facility operations to detect potential hazards to nuclear safety;

h. Performance of special reviews, investigations, or analyses and reports thereon as requested by the Plant Manager - Nuclear or the Chairman of the Company Nuclear Review Board;

~~i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Chairman of the Company Nuclear Review Board;~~ #1

i ~~j.~~ Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Chairman of the Company Nuclear Review Board; E

j ~~k.~~ Review of any accidental, unplanned, or uncontrolled radioactive release including the preparation of reports covering evaluation, recommendations, and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Senior ~~Group~~ Vice President-Nuclear ~~Energy~~ and to the Chairman of the Company Nuclear Review Board; and E E

~~l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.~~ #1

6.5.1.7 The PNSC shall:

~~a. Recommend in writing to the Plant Manager - Nuclear approval or disapproval of items considered under Specification 6.5.1.6a. through d. prior to their implementation;~~ #1

a. ~~b.~~ Render determinations in writing with regard to whether or not each item considered under Specification 6.5.1.6a. through ~~e.~~ d. constitutes an unreviewed safety question; and E

b. ~~c.~~ Provide written notification within 24 hours to the Plant Manager-Nuclear, Senior Vice President-Nuclear and the Company Nuclear Review Board of disagreement between the PNSC and the Plant Manager Nuclear; however, the Plant Manager - Nuclear shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1. E #1

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RECORDS

6.5.1.8 The PNSC shall maintain written minutes of each PNSC meeting that, at a minimum, document the results of all PNSC activities performed under the responsibility provisions of these Technical Specifications. Copies shall be provided to the Senior Vice President-Nuclear and the Company Nuclear Review Board.)#1

ADD INSERT → Ⓐ (P. 6-8A)

6.5.~~2~~3 COMPANY NUCLEAR REVIEW BOARD (CNRB)) Ⓔ

FUNCTION

6.5.~~2~~3.1 The CNRB 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, and
h. Quality assurance practices.

The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

COMPOSITION)#2

ADD INSERT → Ⓑ (P. 6-8B)

~~6.5.2.2 The CNRB shall be composed of the:~~

~~Chairman: Senior Vice President-Nuclear~~
~~Member: Vice President-Nuclear Energy~~
~~Member: Vice President-Engineering Project Management and Construction~~
~~Member: Chief Engineer-Power Plant Engineering~~
~~Member: Director-Quality Assurance~~
~~Member: Director-Nuclear Licensing~~
~~Member: Manager-Power Plant Engineering~~
~~Member: Manager-Nuclear Energy Services~~
~~Member: Manager-Nuclear Fuel~~
~~Member: Group Vice President~~

ALTERNATES

6.5.~~2~~3.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.) Ⓔ

INSERT A

6.5.2 TECHNICAL REVIEW AND CONTROL ACTIVITIES

Technical Department Supervisor

6.5.2.1 The ~~Director, Standards and Technical Support~~ shall assure that each procedure and program required by Specification 6.8 and other procedures which affect nuclear safety, and changes thereto, is prepared by a qualified individual/organization. Each such procedure, and changes thereto, shall be reviewed by an individual/group other than the individual/group which prepared the procedure, or changes thereto, but who may be from the same organization as the individual/group which prepared the procedure, or changes thereto.

6.5.2.2. Individuals responsible for reviews performed in accordance with 6.5.2.1 ~~6.5.2.2, and 6.5.2.3~~ shall be members of the station nuclear ~~production~~ supervisory staff, previously designated by the Plant Manager - Nuclear to perform such reviews. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the appropriate ~~designated~~ review personnel.

6.5.2.3 The station security ~~program~~ plan and implementing procedures shall be reviewed. Recommended changes shall be approved by the ~~Director,~~ Site Services or designated alternate and transmitted to the ~~Vice President-Nuclear Production and to the NSC.~~ CNRB

OPERATIONS SUPERINTENDENT

6.5.2.4 The ~~Director, Standards and Technical Support~~ shall assure the performance of a review by a qualified individual/organization of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering the evaluation, recommendations and disposition of the corrective action to prevent recurrence.

OPERATIONS SUPERINTENDENT

6.5.2.5 The ~~Director, Standards and Technical Support~~ shall assure the performance of a review by a qualified individual/organization of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL. ~~radwaste treatment systems, and the Preplanned Alternate Sampling Program.~~

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COMPOSITION

6.5.~~2~~3.2 The ~~Vice President, Nuclear Operations~~ Executive Vice President shall appoint at least nine members to the ~~NSRG~~ CNRB and shall designate from this membership a Chairman and at least one Vice Chairman. The membership shall collectively possess experience and competence to provide independent review and audit in the areas listed in Section 6.5.2.1 The Chairman and Vice Chairman shall have nuclear background in engineering or operations and shall be capable of determining when to call in experts to assist the ~~NSRG~~ CNRB review of complex problems. All members shall have at least a bachelor's degree in engineering or related sciences or equivalent experience (as per ANS 3.1 Draft Revision 12/6/79, Section 4.1). The Chairman shall have at least ~~10~~ 6 years of professional level management experience in the power field and each of the other members shall have at least 5 years of cumulative professional level experience in one or more of the fields listed in Section 6.5.2.1.

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CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Director to provide expert advice to the CNRB.) E

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per 6 months and as convened by the CNRB chairman or his designated alternate.) E

QUORUM

6.5.2.6 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least four CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.) E

REVIEW

6.5.2.7 The CNRB shall be responsible for the review of:) E

a. The safety evaluations for: (1) changes to procedures, equipment, or systems; and (2) tests or experiments completed under the provision of 10 CFR 50.59, 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 10 CFR 50.59;

c. Proposed tests or experiments which involve an unreviewed safety question as defined in 10 CFR 50.59;

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. All REPORTABLE EVENTS;

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; and

i. Reports and meeting minutes of the PNSC.

PR

AUDITS

6.5.~~2~~3.8 Audits of unit activities shall be performed under the cognizance of the CNRB. These audits shall encompass: E

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months;

b. The performance, training, and qualifications of the entire facility staff at least once per 12 months;

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems, or method of operation that affect nuclear safety, at least once per 6 months;

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months;

e. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel;

f. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year;

g. The Radiological Environmental Monitoring Program and the results thereof at least once per 12 months;

h. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months;

i. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months;

j. The performance of activities required by the Quality Assurance Program for effluent and environmental monitoring at least once per 12 months; and

k. The Emergency Plans and implementing procedures at least once per ~~year;~~ 12 months; E

l. The Security Plans and implementing procedures at least once per ~~year~~; and 12 months

m. Any other area of facility operation considered appropriate by the CNRB or the Executive Vice President.

PR

RECORDS

6.5.~~2~~3.9 Records of CNRB activities shall be prepared, approved, and distributed as indicated below: E

a. Minutes of each CNRB meeting shall be prepared, approved, and forwarded to the Executive Vice President within 14 days following each meeting;

b. Reports of reviews encompassed by Specification 6.5.2.7 shall be prepared, approved, and forwarded to the Executive Vice President within 14 days following completion of the review; and

c. Audit reports encompassed by Specification 6.5.2.8 shall be forwarded to the Executive Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PNSC, and the results of this review shall be submitted to the CNRB, ~~the Vice President of Nuclear Operations,~~ and the Senior Vice President-Nuclear. E

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. In accordance with 10 CFR 50.72, the NRC Operations Center, shall be notified by telephone as soon as practical and in all cases within one hour after the violation has been determined. The Senior Vice President-Nuclear, and the CNRB shall be notified within 24 hours.

b. A Licensee Event Report shall be prepared in accordance with 10 CFR 50.73.

c. The License Event Report shall be submitted to the Commission in accordance with 10 CFR 50.73, and to the CNRB, and the Senior Vice President-Nuclear within 30 days after discovery of the event.

d. Critical operation of the unit shall not be resumed until authorized by the Nuclear Regulatory Commission.

PR

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, Sections 5.1 and 5.3 of ANSI N18.7-1972; and the Facility Fire Protection Program;

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Generic Letter No. 82-33;

c. Security Plan implementation;

d. Emergency Plan implementation;

e. PROCESS CONTROL PROGRAM implementation;

f. OFFSITE DOSE CALCULATION MANUAL implementation; and

g. Quality Control Program ~~Assurance~~ for effluent monitoring using the guidance in Regulatory Guide 1.21, Revision 1, June 1974; and #1

h. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 Each procedure of Specification 6.8.1 (a through g)~~*~~, and changes thereto, shall be reviewed ~~by the PNSC and shall be approved by the Plant Manager - Nuclear~~ prior to implementation and reviewed periodically as set forth in administrative procedures. in accordance with Specification 6.5.1 and 6.5.2, as applicable, #2 #4

6.8.3 Temporary changes to procedures ~~of Specification 6.8.1 (a through g)*~~ may be made provided: #2

a. The intent of the original procedure is not altered;

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Operator license on the unit affected; and

c. The change is documented, reviewed ~~by the PNSC,~~ and approved ~~by the Plant Manager - Nuclear~~ within 14 days of implementation. #4

in accordance with 6.5.1 and 6.5.2, as applicable,

~~*The Quality Control Program and procedures for environmental monitoring (6.8.1.h) are reviewed by the State of Florida.~~ #2

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PR

PROCEDURES AND PROGRAMS (Continued)

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 Safety Injection System, Chemical and Volume Control System, and the Containment Spray System.~~ The program shall include the following: #3

(1) Preventive maintenance and periodic visual inspection requirements, and

(2) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(1) Training of personnel,

(2) Procedures for monitoring, and

(3) Provisions for maintenance of sampling and analysis equipment.

c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(1) Identification of a sampling schedule for the critical variables and control points for these variables,

(2) Identification of the procedures used to measure the values of the critical variables,

(3) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(4) Procedures for the recording and management of data,

PR

PROCEDURES AND PROGRAMS (Continued)

(5) Procedures defining corrective actions for all off-control point chemistry conditions, and

(6) A procedure identifying: (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-Accident 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:

(1) Training of personnel,

(2) Procedures for sampling and analysis, and

(3) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, DC pursuant to 10 CFR 50.4.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following: (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

~~The initial Startup Report shall address each of the startup tests identified in Chapter 13 of the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.~~ Subsequent Startup Reports shall address startup tests that are necessary to demonstrate the acceptability of changes and/or modifications. #1

REPLACE WITH ATTACHED INSERT

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, except the Quality Control Program for environmental monitoring, shall be reviewed by the PNSC and approved by the Plant Manager - Nuclear prior to implementation and periodically as provided by procedure.

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 Operators License on the unit affected.

c. The change is documented, reviewed by the PNSC and approved by the Plant Manager - Nuclear within fourteen days of implementation.

6.9 REPORTING REQUIREMENTS

In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following identified reports shall be submitted to the U. S. Nuclear Regulatory Commission, Document Control Desk, Washington DC. pursuant to 10 CFR 50.4.

6.9.1 ROUTINE REPORTS

a. Startup Report - 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. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions of 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.

USE THIS AS INSERT TO 6.9.1.1

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

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STARTUP REPORT (Continued)

Startup Reports shall be submitted within: (1) 90 days following completion of the Startup Test Program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of Startup Test Program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL REPORTS*

6.9.1.2 Annual Reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. ~~The initial report shall be submitted prior to March 1 of the year following initial criticality.~~ E

Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions** (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, thermoluminescent dosimeter (TLD), or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole-body dose received from external sources should be assigned to specific major work functions;

b. The results of specific activity analyses in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded (in graphic and tabular format); (2) 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 degassing 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.

*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.

**This tabulation supplements the requirements of § 20.407 of 10 CFR Part 20.

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ANNUAL REPORTS (Continued)

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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, with operational controls, as appropriate, and with 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 the Land Use Census required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the Offsite Dose Calculation Manual, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the Radiological Environmental Monitoring Program; at least two legible maps** covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results of licensee participation in the Interlaboratory Comparison Program and the corrective action taken if the specified program is not being performed as required by Specification 3.12.3; reasons for not conducting the Radiological Environmental Monitoring Program as required by specification 3.12.1, and discussion of all deviations from the sampling schedule of Table 3.12-1; discussion of environmental sample measurements that exceed the reporting levels of Table 3.12-2 but are not the result of plant effluents, pursuant to ACTION b. of Specification 3.12.1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

E

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT *

6.9.1.3 Routine Annual Radiological Environmental Operating Reports covering the operation of the facility during the previous calender year shall be submitted prior to May 1 of each year.

*A single submittal may be made for a multiple unit station.

** One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

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SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.4 Routine Semiannual 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.~~ E

The Semiannual 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. For solid wastes, the format for Table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR Part 61), type of container (e.g., ~~LSA,~~ Type A, Type B, ~~Large Quantity~~) and SOLIDIFICATION agent or absorbent (e.g., cement, ~~urea formaldehyde~~ strong tight package). #5

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), 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 (Figure 5.1-1) 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. Approximate and conservative ~~approximate~~ methods may be used in lieu of actual meteorological measurements. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM). E #2

~~*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.~~

*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. E

**In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases from the previous calendar year. ~~and other nearby uranium fuel cycle sources, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR Part 190, "Environmental Radiation Protection Standards for Nuclear Power Operation."~~ Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, March 1976. #3

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Semiannual Radioactive Effluent Release Reports (and) shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.3.6 or 3.3.3.7, respectively; description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Specification 3.11.1.4 or 3.11.2.6~~; an explanation for the unavailability of vegetation samples and justification for selecting a new location as specified in 3.12.1; and an explanation of a new land use location with information supporting the changed sample location as specified in 3.1.2.2.~~ E #6

MONTHLY OPERATING REPORTS

6.9.1.5 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 U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

~~RADIAL~~ PEAKING FACTOR LIMIT REPORT (Replace with Insert (P. 6-18A)) #4

6.9.1.6 The F_{xy} ~~limits~~ for RATED THERMAL POWER (F_{xy}^{RTP}) ~~shall be~~ established on at least each reload core and ~~shall be maintained~~ available in the Control Room. The ~~limits shall~~ be established and implemented on ~~a time scale consistent with normal procedural changes.~~

INSERT TO PAGE 6-18

6.9.1.6 The W(Z) function(s) for Base-Load Operation corresponding to a ±2% band about the target flux difference and/or a ±3% band about the target flux difference, the Load-Follow function $F_Z(Z)$ and the augmented surveillance turnon power fraction, P_T, shall be provided to the U.S. Nuclear Regulatory Commission, whenever P_T is < 1.0. In the event, the option of Baseload Operation (as defined in Section 3.2.6.a [3]) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_Z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.

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~~RADIAL~~ PEAKING FACTOR LIMIT REPORT (Continued) #4

The analytical methods used to generate the ~~F_{xy}~~ Peaking Factor limits shall be those previously reviewed and approved by the NRC.** If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

~~A report containing the F_{xy} limits for all core planes containing Bank "D" control rods and all unrodded core planes along with the plot of predicted ($F_Q^T \cdot P_{Rel}$) vs axial core height (with the limit envelope for comparison) shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation.~~ #4

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the U.S. Nuclear Regulatory Commission, Document Control Desk, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, ~~or~~ 4.0, or 5.0. #7 E

6.10 RECORD RETENTION

6.10.1 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.

6.10.2 The following records shall be retained for at least 5 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 EVENTS;

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; and

h. Records of annual physical inventory of all sealed source material of record.

~~*WCAP-8385 "Power Distribution Control and Load Following Procedures" and WCAP-9272-A "Westinghouse Reload Safety Evaluation Methodology."~~ #4

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RECORD RETENTION (Continued)

facility

6.10.3 The following records shall be retained for the duration of the ~~unit~~ Operating License: (E)

a. Records and drawing changes reflecting ~~unit~~ facility design modifications made to systems and equipment described in the Final Safety Analysis Report;

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories;

c. Records of facility radiation and contamination surveys;

d. Records of radiation exposure for all individuals entering radiation control areas;

e. Records of gaseous and liquid radioactive material released to the environs;

f. Records of transient or operational cycles for those unit components identified in Table 5.7-1;

g. Records of reactor tests and experiments;

h. Records of training and qualification for current members of the facility staff;

i. Records of inservice inspections performed pursuant to these Technical Specifications;

j. Records of quality assurance activities required for the duration of the ~~unit~~ facility Operating License by the Quality Assurance Manual; (E)

k. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59;

l. Records of meetings of the PNSC and the CNRB;

m. Records of the service lives of all hydraulic and mechanical snubbers required by Specification 3.7.~~X~~6 including the date at which the service life commences and associated installation and maintenance records; (E)

n. Records of secondary water sampling and water quality; and

Replace with Insert (P.6-20A)

o. ~~Records of analyses required by the Radiological Environmental Monitoring Program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and QA records showing that these procedures were followed.~~ #1

k. Records of meetings of the PNSC and the CNRB.

l. Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m. Records of the service lives of all snubbers required by specification 3.13 including the date of which the service life commences and associated installation and maintenance records.

Insert to P. 6-20

~~n.~~ o. Annual Radiological Environmental Monitoring Reports and records of analyses transmitted to the licensee which are used to prepare the Annual Radiological Environmental Monitoring Report.

#1

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. 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 issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. Each High Radiation Area in which the intensity of radiation is greater than 1000 mRem/hr shall be subject to the provisions of 6.12.1(a) above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under administrative control.

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RECORD RETENTION (Continued)

p. Records for Environmental Qualification which are covered under the provisions of 10 CFR 50.49.

6.11 RADIATION PROTECTION PROGRAM

6.11.1 Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 Pursuant to paragraph 20.203(c)(5) of 10 CFR Part 20, in lieu of the "control device" or "alarm signal" required by paragraph 20.203(c), each high radiation area, as defined in 10 CFR Part 20, in which the intensity of radiation is equal to or less than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates 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). Individuals qualified in radiation protection procedures (e.g., Health Physics Technician) or personnel continuously escorted by such individuals may be exempt from the RWP issuance requirement during the performance of their assigned duties in high radiation areas with exposure rates equal to or less than 1000 mR/h, provided they are otherwise following plant radiation protection procedures for entry into such high radiation areas. 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; or

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 levels in the area have been established and personnel have been made knowledgeable of them; or

c. An individual 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 Health Physics Shift Supervisor in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mR/h at 45 cm (18 in.) from the radiation source or from any surface which the radiation penetrates shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the shift

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HIGH RADIATION AREA (Continued)

~~Foreman~~ supervisor on duty and/or health physics ~~supervision.~~ personnel 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 areas and the maximum allowable stay time for individuals in that area. In lieu of the stay time specification of the RWP, direct or remote (such as closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities being performed within the area. E

For individual high radiation areas accessible to personnel with radiation levels of greater than 1000 mR/h that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual area, that individual area shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be ~~approved~~ reviewed by the ~~Commission~~ PNSC prior to implementation. #1

6.13.2 Licensee-initiated changes to the PCP:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

(1) Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

(2) A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

(3) Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

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:

a. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

PR

OFFSITE DOSE CALCULATION MANUAL (ODCM) (Continued)

(1) 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, dated and containing the revision number, together with appropriate analyses or evaluations justifying the change(s);

(2) A determination that the change will not reduce the accuracy or reliability of dose calculations or Setpoint determinations; and

(3) Documentation of the fact that the change has been reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

~~6.15 MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS*~~ #1

~~6.15.1 Licensee-initiated major changes to the Radwaste Treatment Systems (liquid, gaseous, and solid):~~

~~a. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PNSC. The discussion of each change shall contain:~~

~~(1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;~~

~~(2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;~~

~~(3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems;~~

~~(4) 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;~~

~~(5) An evaluation of the change, which shows the expected maximum exposures to a MEMBER OF THE PUBLIC in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the License application and amendments thereto;~~

~~(6) 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 change is to be made;~~

*Licensees may choose to submit the information called for in this Specification as part of the annual FSAR update.

PR

MAJOR CHANGES TO LIQUID, GASEOUS, AND SOLID RADWASTE TREATMENT SYSTEMS (Continued)

(7) An estimate of the exposure to plant operating personnel as a result of the change; and

(8) Documentation of the fact that the change was reviewed and found acceptable by the PNSC.

b. Shall become effective upon review and acceptance by the PNSC.

T. S. Number 1.0

Justification:

1. The new RPI scales take into account the non-linear response of the RPI's that would otherwise be accounted for by the Reference Position. The new scales allow the RCCO to directly compare the RPI position to the step counter position without having to refer to the Reference Position curve. Removal of the "Reference Position" alleviates some of the unnecessary workload on the RCCO.

2. K_{eff}=0.95 is consistent with Amendment 132/126 (7/18/88).

3. A CONTROLLED LEAKAGE definition has been added to complement the addition of CONTROLLED LEAKAGE restrictions added to Specification 3.4.6.2.

4. All of our shipments do not go directly to a burial site, some are shipped to a waste processor for volume reduction prior to burial.

5. A limit of 24 months on "refueling" surveillances is a management action to limit time between surveillances.

T. S. 2.1.2

Justifications:

1. BASES - Per FSAR Table 4.1-9, the reactor vessel and pressurizer are designed to the requirements of ASME Section III.

T.S. Number 2.2.1

Justifications:

1. The CTS does not address "allowable" values. The 1986 version of RTS adopted the STS 2-column format containing nominal setpoints and allowables. The RTS allowables were best estimate numbers determined by Westinghouse using generic tolerances. However, FPL would require additional analysis to adequately support these values.

2. Pzr. pressure: No credit is taken for PORVs in the safety analysis or the basis in the CTS. (PORV functional capability enhances the overall reliability of the RCS).

3. Turbine trip: P-7 blocks reactor trip on turbine trip.

4. Items 17.b(1) and 17.d are confusing. P-10 in 17.d, which allows a block of PR trip during plant startup, should be greater than or equal to 10%. P-10 reset point in 17.b(1), which allows a block of at-power trips during shutdown, should be less than or equal to 10%. Revising the wording to refer to 17.b(1) as the reset of the P-10 bistable and 17.d as the tripping of the bistable will clarify the tech spec and better represent instrument design.

T. S. 3.0.3

Justifications:

1. Flexibility required for dual unit shutdown to prevent a severe transient on the Florida electrical grid. This comment is consistent with previous discussions and agreements during RTS working meetings regarding dual unit shutdown due to electrical components being inoperable. (Also applies to BASES change.)

T. S. 3.0.6

Justification:

1. This comment provides a minor revision, for clarification only, of CTS 3.0.5. This current specification was approved for inclusion in the CTS in Amendment 114/108 dated June 27, 1985.

   The rewritten portion of the comment closely matches the wording of the NRC letter dated April 10, 1980 to All Power Reactor Licensees concerning the definition of OPERABLE.

   Although this tech spec is not currently in the desk reference STS, FPL feels that it is important to include this clarification.

   (This justification also applies to BASES change.)

T. S. Number 4.0.3 BASES

Justifications:

1. This change clarifies that the allowable outage time in the ACTION Statement begins at the end of the 24 hour allowance.

T. S. Number 4.0.6 BASES

Justifications:

1. This change was made to eliminate the need for unnecessary duplicate testing.

T. S. Number 3/4.1.1.1

Justifications:

1. BASES - Revised to reflect actual condition addressed by the LCO. "Postulated steam line break" appears inappropriate with RCS temperature below 200F.

2. Figure 3.1-1: Breakpoints should be shown for operator clarity.

3. During the first 60 EFPD of a cycle, it is possible that an adjustment to the letdown curve may not have to be made if there is good agreement between measured data and design figures.

T. S. Number 3/4.1.1.2

Justifications:

1. The term "equivalent" would mean 100+ pgm of 1950 ppm; this would require 2 charging pumps in Mode 5. No rate requirements for boration in Mode 5.

T. S. Number 3/4.1.1.3

Justifications:

1. Consistency on reporting requirements. This is a generic issue on 30 day reporting.

T. S. Number 3/4.1.2.1

Justifications:

1. STS requires temperature verification once per 7 days. There is no CTS requirement to verify temperature, hence, 7 days is more restrictive than CTS.

2. Certain areas of the plant may be inaccessible due to radiological considerations.

T. S. Number 3/4.1.2.2

Justifications:

1. STS requires temperature verification once per 7 days. There is no CTS requirement to verify temperature, hence, 7 days is more restrictive than CTS.

2. Certain areas of the plant may be inaccessible due to radiological considerations.

T. S. Number 3/4.1.2.4

Justifications:

1. STS requires temperature verification once per 7 days. There is no CTS requirement to verify temperature, hence 7 days is more restrictive than CTS.

T. S. Number 3/4.1.2.5

Justifications:

1. STS requires temperature verification once per 7 days. There is no CTS requirement to verify temperature, hence, 7 days is more restrictive than CTS.

2. 3/4.1.2 BASES clarified to indicate background for RWST temperature; for operator information.

T. S. Number 3/4.1.2.6

Justifications:

1. This change is consistent with related tech specs (i.e., 3.1.2.1, 3.1.2.2 and 3.1.2.5). There is no CTS requirement to verify temperature.

T. S. Number 3/4.1.3.1

Justifications:

1. The new RPI scales take into account the non-linear response of the RPI's that would otherwise by accounted for by the Reference Position. The new scales allow the reactor operator to directly compare the RPI position to the step counter postion without having to refer to the Reference Position curve. Removal of the "Reference Position" alleviates unnecessary workload on the reactor operator. (This justification also applies to BASES change.)

T. S. Number 3/4.1.3.2

Justifications:

1. Revised for consistency with CTS 3.2.2 and related bases: "Reduction in power to 75% (3 loop) . . . will ensure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions." Power reduction for an indicated mispositioned rod should be the same as a known mispositioned rod. (Tech. Spec. 3.1.3.1 ACTION c.3.a.)

2. Delete reference to Reference Position. Refer to justification to 3/4.1.3.1 for further information.

T. S. Number 3/4.1.3.4

Justifications:

1. This surveillance will require a unit shutdown to be performed. Surveillance 4.1.3.4a and b cover any maintenance that could affect the rod drop times.

T. S. Number 3/4.1.3.5

Justifications:

1. Comment returns surveillance to STS wording which is appropriate for Turkey Point.

T. S. Number 3/4.1.3.6

Justifications:

1. Comment returns surveillance to STS wording which is appropriate for Turkey Point.

T. S. Number 3/4.2.1

Justifications:

1. Performing surveillance for greater than a 6 hour period provides negligible information. This comment was agreed to by NRC technical reviewer on April 7, 1988.

   Adding a functional test will be consistent with other surveillances in STS and meets the intent of the surveillance.

T. S. Number 3/4.2.2

Justifications:

1. The NRC proposed STS wording would require FPL to perform additional analysis to generate the Peaking Factor Limit Report per T.S. 6.9.1.6. In addition, the NRC's proposed wording deleted the Augmented Surveillance Tech. Specs. which were approved by NRC on March 17, 1982, as License Amendment 80/74. The Augmented Surveillance Tech. Specs. were developed as a contingency against reductions in the F_Q margin. FPL has experienced F_Q margin reductions due to steam generator tube plugging, changes in analytical methods, and unexpected errors in design codes. Equipment currently in place to implement the Augmented Surveillance Tech. Spec. would have to be removed if this surveillance is discontinued.

T. S. Number 3/4.2.4

Justifications:

1. There is no requirement in the current tech specs to verify the excore QPTR calculation with an incore flux map. Monitoring the incore tilt with a flux map once per 24 hours will be significantly more conservative than the current tech spec.

   With one power range detector out of service, the remaining power range detectors along with other control room instrumentation (i.e., incore T/C, Tavg, flows, etc.) will detect changes in core parameters.

2. Several plants currently use incore thermocouples to determine QPTR. Incore thermocouples provide a rapid means of determining the core relative radial power distribution for use on an on-line basis. Thermocouples are currently used in QSPDS. When used in conjunction with moveable detector map, incore thermocouples can be normalized to provide accurate relative integrated fuel assembly power distribution measurements.

T. S. Number 3/4.2.5

Justifications:

1. The present equipment at Turkey Point does not allow performing this surveillance without a plant shutdown and containment entry behind the biological shield wall.

T. S. <u>3/4.3.1</u>

Justifications:

1. This change reflects the Plant design as shown in attached FSAR Figure 7.2-5. These trips do not apply below P-7.

2. Provides clarification to operators on required safety function based on increasing and decreasing power, i.e., automatically enabling trips, etc.

3. This surveillance requires Plant to be in a shutdown condition. An 18-month surveillance frequency may cause FPL to have to shutdown for refueling early or cause an unnecessary outage.

4. This surveillance has been revised to create consistency between the table and the table notations.

5. There is no LCO which corresponds to this surveillance and the trip bypass breaker circuit is already tested by Note 13.

6. The Automatic Trip and Interlock Logic is not testable prior to mode 3; therefore, the surveillance requirement has been removed.

7. Use of jumpers/fuses to place channel in trip condition was agreed to by NRC technical reviewer in FPL/NRC working meetings.

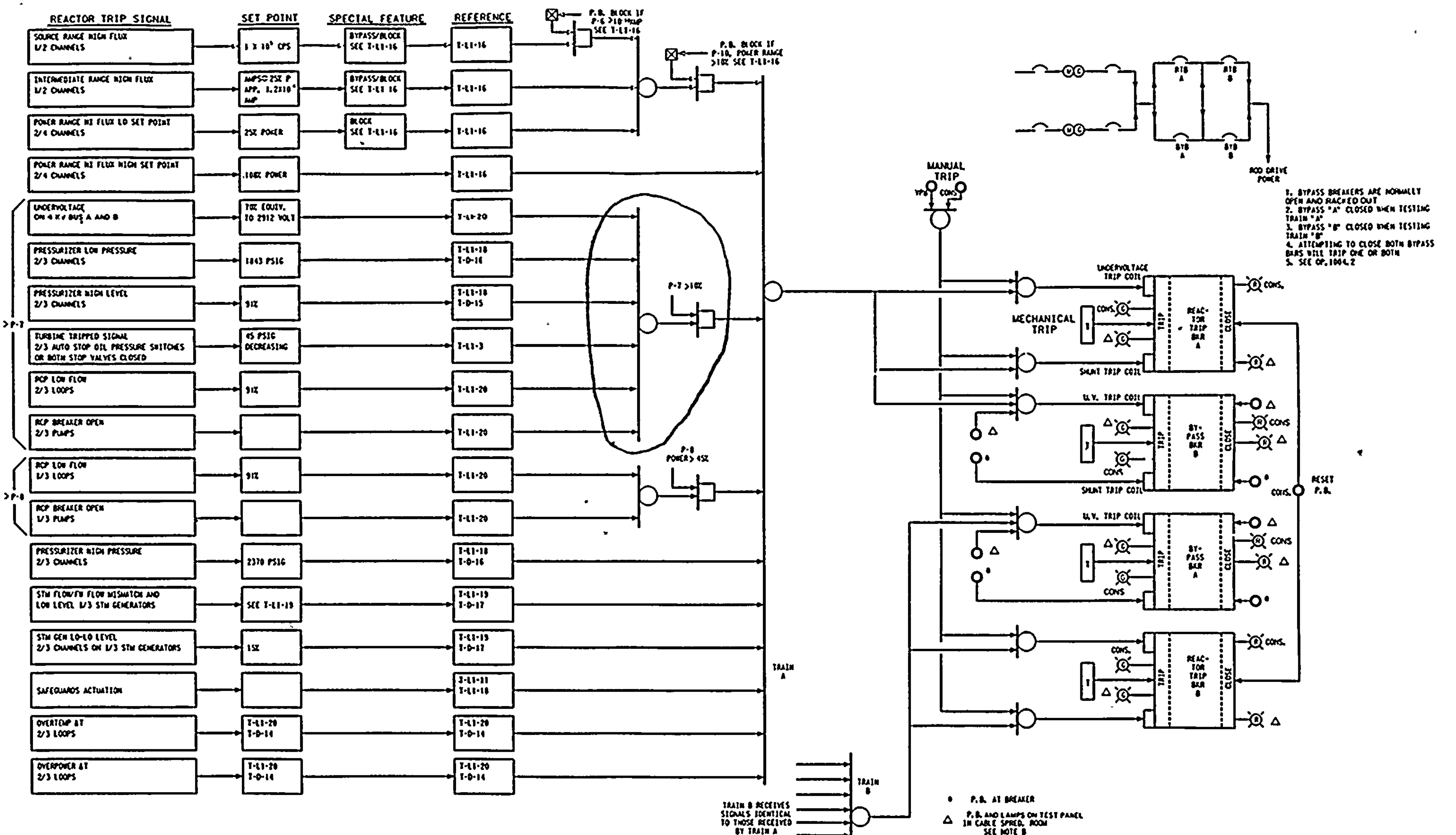
8. STS wording is adequate to give guidance for 30-75% RTP.

9. Above P-7 and below P-8, 2 RCP "breakers open" are required to cause a reactor trip. Based upon Plant design, a single underfrequency channel/bus may not lead to a RCP breaker/reactor trip. A requirement for 2/bus will lead to a reactor trip due to an underfrequency on either bus.

10. The P-10 logic circuitry provides a redundant signal.

11. In the decreasing power mode, two (2) channels to trip are required (see Justification #2 above).

12. In the decreasing power mode, three (3) channels to trip are required (see Justification #2 above).



NOTE A

A TRIP SIGNAL TO THE UNDERVOLTAGE TRIP COIL DE-ENERGIZES THE COIL DROPPING OUT THE LATCH AND FREEING THE TRIP SPRING TO TRIP THE BREAKER.
A TRIP SIGNAL TO THE SHUNT TRIP COIL ENERGIZES THE SHUNT TRIP COIL. MAGNETIC FORCE PULLS THE LATCH FREE OF THE TRIP SPRING.

NOTE B

• P.B.'S CLOSE / TRIP AT BKR ACTIVE ONLY WITH BYPASS BKR IN TEST POSITION

△ P.B. CLOSE AT TEST PANEL ACTIVE WITH BKR RACKED IN.

△ P.B. TRIP AT T.P. ACTIVE WITH BKR RACKED IN OR OUT.

Rev 4 7/86

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT PLANT UNITS 3 & 4

REACTOR TRIP SIGNALS

FIGURE 7.2-5

REF DWG: 5610-T-LI SH. 2 (REV. 8)

desires to reserve the option to cock the rods (shutdown banks) out of the core during Mode 3 for operational flexibility.

To address the issue, Mr. Rick Mande has indicated that FPL would like to institute an administrative control during Mode 3 that would require the rod control system to be either 1) placed in the "bank select" mode with a shutdown bank as the bank selected, or 2) disabled by opening the reactor trip breakers. Westinghouse agrees that this is an acceptable means of preventing uncontrolled rod withdrawal during Mode 3, and is therefore an effective means of addressing the issue on a short-term basis.

Please call me if you have any questions.

Very truly yours,

B. A. Pearson

B.A. Pearson
Project Engineer

BAP:kph

cc: R.J. Acosta
D.C. Bradford
S.G. Brain
W.E. Coe
R.A. Decker
J.A. DeMastry
T.G. Grozan
D.W. Haase
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S.H. Shepherd
A.E. Siebe
F.H. Southworth
C. Villard
R.W. Winnard
S.K. Mathavan
C.G. Smallwood

'83 11:22 ... 03 16:4...

3/4.4.1.2

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Services Integration Division

Box 2728
Pittsburgh Pennsylvania 15230 2728

August 15, 1984
FPL-84-729

Mr. K. N. Harris, Vice President
Turkey Point Nuclear
Florida Power and Light Company
P. O. Box 029100
Miami, Florida 33152

Dear Mr. Moaba:

Florida Power & Light Company
Turkey Point Units 3 and 4
UPDATE - CONSISTENCY BETWEEN SAFETY ANALYSIS
AND TECHNICAL SPECIFICATIONS CONCERNING
NUMBER OF REACTOR COOLANT PUMPS IN OPERATION

In June of this year you were notified of a potential unreviewed safety question concerning an inconsistency between the safety analysis and the Tech Specs. This inconsistency applies to the number of operating reactor coolant pumps when in Mode 3 (or the equivalent) of the Tech Specs.

Since that time, Westinghouse has met with the NRC staff at their request to present the Westinghouse position on this issue and recommendations for resolution. The material presented at this meeting is documented in Westinghouse letter NS-EPR-2935, which is attached for your information (cover letter only).

The purpose of this letter is to update you on the latest information concerning this issue. Westinghouse will notify you of further developments as they occur.

If you have any questions, please contact me.

Very truly yours,

D. J. Richards, Manager
Projects Department
South Area

HT/413L
Attachment

cc: W. H. Rogers, Jr.
K. N. Harris
H. D. Mantz
P. P. DeRosa
H. E. Yaeger
C. J. Baker
H. N. Paduano
S. G. Brain
D. J. Richards
T. F. Sullivan
E. C. Anderson W
E. V. Rutledge W

3/4,4.1.2

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Technology Division

Box 355
Pittsburgh Pennsylvania 15230

July 9, 1984
NS-TA-84-003

Mr. D. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
2920 Norfolk Avenue
Washington, D.C. 20555

Dear Mr. Eisenhut:

NUMBER OF OPERATING REACTOR COOLANT PUMPS IN MODE 3

This letter formalizes the material presented on June 15, 1984, with respect to the consistency between the Technical Specifications and the safety analysis for the number of operating reactor coolant pumps in Mode 3. This meeting was held at the request of the NRC staff in order to discuss the Westinghouse determination of a potential unreviewed safety question for three and four loop plants for this issue. Enclosed are ten (10) proprietary copies of the slides and ten (10) non-proprietary copies. Also enclosed are one (1) copy of Application for Withholding, AW-84-63 (non-proprietary) and one (1) copy of Affidavit (non-proprietary).

As part of an informal review of a utility's Tech Specs by the NRC Reactor Systems Branch, the staff asked what the safety analysis assumptions were concerning the number of operating reactor coolant pumps, particularly at or near zero power. Although the question was never formally asked, Westinghouse reviewed the analysis assumptions with respect to the Tech Specs.

The requirement for operating reactor coolant pumps under these conditions is contained in Specification 3.4.1.2 of the Standard Tech Specs. In non-Standard Tech Specs, the requirement is contained in Specification 3.1. These Specs state that when the plant is subcritical by the shutdown margin between 350°F (RHR cut-in) and 547°F or 557°F (no-load conditions), there must be two loops operable, but only one loop has to be actually operating.

However, the safety analysis in the FSARs assumes that either two or all of the reactor coolant pumps are operating, not just one. (At the staff's request, the assumptions made concerning the number of operating pumps have been noted for those plants within Westinghouse scope in the attachment). The accidents which are limiting at zero power are steamline break, rod ejection, and bank withdrawal from subcritical. Westinghouse has reviewed these accidents under the reduced flow conditions of one pump. For the rod ejection and steamline break events, Westinghouse has determined that the inconsistency between the safety analysis

and the Tech Spec will not impact the conclusions presented in the FSAR. For the bank withdrawal from subcritical event, Westinghouse has performed calculations which show that the DNB design basis may not be met when only one pump is in operation. Thus, the margin of safety as defined in the basis of the Tech Specs is reduced.

Westinghouse has also performed calculations for one pump operation assuming more realistic, but still conservative, reactivity insertion rates. The results of these calculations show that the DNB design basis is met. Other assumptions and models used in these analyses are identical to the FSAR methods of analysis for this event. Thus, Westinghouse feels that no significant safety hazard exists.

Westinghouse is currently considering long term analytical solutions to this issue which will show that the DNB design basis can be met when only one reactor coolant pump is in operation so that the Tech Specs will not need to be changed. However, in the short term, Westinghouse recommends that the plants be operated with the same number of reactor coolant pumps in operation as was assumed in the analysis. Note that this is not a realistic requirement when the plant is cooling down prior to going into Mode 4 (RHR operation), particularly for those plants for which the analysis assumes all pumps in operation. Thus, an alternative to having more than one pump in operation is to prevent rod withdrawal. This will preclude the accident from taking place. Although physical prevention of withdrawal will accomplish this, administrative procedures may be preferable. The ability to cock the rods partway out of the core during Mode 3 provides desired operating flexibility. Furthermore, there is no mechanism by which the control rods can be automatically withdrawn in Mode 3 due to a control system error. Increased operator awareness during this time and adherence to procedures will also prevent the accident from occurring.

Finally, while Westinghouse feels that it is appropriate to consider bank withdrawal when in Mode 3, Westinghouse does not intend to address this event in other modes of operation (Standard Tech Spec Modes 4 and 5). Bank withdrawal from subcritical is a valid scenario when going from Mode 3 to Mode 2. However, consideration of bank withdrawal in Modes 4 and 5 is unrealistic and it is questionable as to whether it is applicable or if it is a Condition II event. Again, increased operator awareness must be considered when evaluating the appropriateness of the event.

Correspondence with respect to the Westinghouse affidavit or application for withholding should reference AW-84-63, and should be addressed to Mr. R. A. Wiesemann, Manager, Regulatory and Legislative Affairs, P.O. Box 355, Pittsburgh, Pennsylvania 15230. Other correspondence or questions should be directed to Mr. J. L. Little, Manager, Operating Plant Licensing Support, 412/374-5054.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

E. P. Rahe, Jr.
Nuclear Safety Department

M. P. Osborne/ds

Enclosures

[illegible]-RAT-PTA-84-089

3/4.4.1-2

Nuclear Safety Department
FROM: Risk Assessment Technology
WIN: 284-4303
DATE: October 4, 1984
SUBJECT: FPL/FLA Prevention of Rod Withdrawal in Mode 3

OCT [illegible]

To: B. A. Pearson MMOB 2-17

cc: M. P. Osborne MNC 4-09A
P. A. Loftus MNC 4-09A
P. W. Robertson MNC 4-09A
M. N. Raymond MNC 4-09A

Ref: 1. FPL-84-729, 8-15-84
2. NS-EPR-2935, 7-9-84
3. FP-FP-697, 82FP*-G-011

Florida Power and Light has requested additional information regarding an inconsistency between the safety analysis and the Tech Specs with respect to the number of operating reactor coolant pumps when in Mode 3 (or the equivalent) as defined in the Tech Specs. Reference 1 provided the latest update on this issue. The purpose of this letter is to provide confirmation of the safety analysis assumptions and FP&L's proposed administrative procedure to address this issue, as discussed with Rick Mende, Reactor Engineer at Turkey Point.

As discussed in Reference 2, a letter from Westinghouse to the NRC, the only event impacted by this issue is the RCCA bank withdrawal from subcritical. This accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of RCCAs resulting in a power excursion. Such a transient could be caused by a malfunction of the automatic rod control system or by operator error. The analysis assumes a reactivity insertion rate of 75 pcm/sec, greater than the maximum resulting from the simultaneous withdrawal of the combination of two sequential control banks having the maximum combined worth at maximum speed.

The most recent RCCA bank withdrawal from subcritical analysis for Turkey Point Units 3 & 4 was performed as part of the positive MTC study in 1981 (Reference 3), and assumes that all three reactor coolant pumps are operating. However, the Turkey Point Tech Specs state that "In Hot Shutdown at least two Reactor Coolant Loops shall be operable and at least one Reactor Coolant Loop shall be in operation." (T.S. 3.4.1.d, pg 3.4-[illegible]) Thus, the Tech Specs are inconsistent with the analysis assumption. As [illegible] in Reference 2, it is not a realistic requirement to have all three reactor coolant pumps operating when the plant is cooling down prior to going to Cold Shutdown. In addition, administrative procedures are preferable to physical prevention of rod withdrawal for Turkey Point, since the plant desires to reserve the option to cock the rods (shutdown banks) out of the core during Mode 3 for operational flexibility.

address the issue, Rick Mende has indicated that FP&L would like to institute an administrative control during Mode 3 that would require the rod control system to be either 1) placed in the "bank select" mode with a shutdown bank as the bank selected, or 2) disabled by opening the reactor trip breakers. Westinghouse agrees that this is an acceptable means of preventing uncontrolled rod withdrawal during Mode 3, and is therefore an effective means of addressing the issue on a short-term basis.

The foregoing has been telecopied to FP&L, and their comments have been incorporated. Please formally transmit this information to the customer. Contact me if there are any questions.

Glenn Heberle

G. H. Heberle
Plant Transient Analysis

T. S. 3/4.4.1.3

Justifications:

1. Allowing the standby RHR loop to be inoperable for surveillance testing would cause only one loop to be OPERABLE for the two-hour period of time allowed for performing the tests. The potential exists that the remaining RHR pump could be de-energized for one (1) hour (as allowed by another footnote) or that it would otherwise become inoperable, resulting in having no operating loop for the specified time period. This does not represent a significant increase in risk because the RCS thermal capacity is sufficient to maintain the RCS temperature rise within acceptable limits during this time period, while the RHR loop being surveilled is restored to its OPERABLE state.

T. S. 3/4.4.2

Justifications:

1. BASES - The relief capacity of the pressurizer safety valves is 293,300 lbs/hr per FSAR Table 4.1.3

T. S. <u>3/4.4.2.2</u>

Justifications:

1. This revised action statement allows cooldown of a unit when safety valves are declared inoperable to perform maintenance. (Cooldown is considered a positive reactivity change.)

2. The definition of RCS pressure boundary is consistent with OMS protection that provides at least 2.20 square inches to depressurize the RCS.

T. S. <u>3/4.4.3</u>

Justifications:

1. BASES - The maximum pressurizer water volume at 92% indiated water level is 1133 cubic feet.

2. This surveillance should not be performed during power operation.

T. S. 3/4.4.4

Justifications:

1. A revised specification has been proposed which covers only the PORV block valves.

   Reactor Coolant System overpressure protection is provided by the Pressurizer Safety Valves as addressed in Specification 3/4.4.2.

   The Steam Generator tube rupture accident does require a means to depressurize the Reactor Coolant System to reduce coolant leakage to the Secondary Side of the Steam Generator. The primary means of depressurizing the primary system is by use of the normal pressurizer spray. Auxiliary pressurizer sprays can be used as a backup. While the PORV can be used as a second backup, it is the least desirable because it tends to reduce Reactor Coolant System inventory.

2. ACTION has been revised to indicate HOT SHUTDOWN (Mode 4) consistent with the APPLICABILITY (Modes 1-3).

T. S. 3/4.4.5

Justifications:

1. Specification 3.0.4 provides restriction for MODE change. Shutdown requirements are provided because Specification 3.0.3 would apply if a steam generator becomes inoperable during operation. Current wording is confusing to operator because it does not provide all required actions.

2. The intent of this surveillance is to inspect every refueling interval.

3. Consistency on reporting with 10 CFR 50.73 for LER's.

T. S. 3/4.4.6.1

Justifications:

1. The CTS requirement is refueling (recently approved amendment). Surveillance cannot be performed with unit at power or greater than 200°F due to ALARA and HP restrictions on dose rates and stay times.

T. S. 3.4.4.6.2

Justifications:

1. CONTROLLED LEAKAGE requirements were added to explicitly control flow from the RCP seals. The wording is similar to the STS.

2. This LCO requirement was revised to be consistent with CTS (NRC Order, dated 4/20/81) restrictions for valve leakage, which are explicitly called out in Table 3.4-1.

3. The added footnote is from CTS 4.17, Reactor Coolant System Pressure Isolation Valves. It allows flexibility in measurement of valve leakage in order to reduce personnel radiation exposure.

4. Consistent with CTS and 3.0.3.

5. This surveillance was added due to the addition of an LCO for CONTROLLED LEAKAGE. The wording is similar to the STS.

6. The added wording excludes the requirement to verify valve leakage if work performed on the valve would not reasonably be expected to affect valve leakage; e.g., painting.

7. BASES - Wording was added to describe CONTROLLED LEAKAGE. The STS words describing safety injection flow were not included, as they were not applicable to Turkey Point.

8. BASES - Leakage from the RCS pressure isolation valves is usually isolated during normal operation due to various system line ups. Wording was added to clarify that only actual mass loss is considered in calculating the allowed limit of IDENTIFIED LEAKAGE.

9. A surveillance is added to require testing of the check valves after actuation (flow through the valve), similar to the STS.

   This testing is required prior to entering Mode 2 because operating pressure is required to conduct the test.

T. S. Number 3/4.4.7

Justifications:

1. A representative sample of the RCS can not be obtained in this condition. This footnote is consistent with recent FPL/NRC discussions at Turkey Point.

T. S. 3/4.4.8

Justifications:

1. BASES - Due to the allowance given for transport of releases to site boundary (30 min.) and two (2) hours from sample counting, isotopes with halflives, <10 min would decay through 12 halflives which would make the data very inaccurate even if these isotopes were still detectable. Industry assumptions are, that after 8 half lifes, the isotope is gone, i.e., <1% of isotopes original activity remains, and after 120 min. (sampling allowance), 12 halflives for an isotope with a 10 min. half life, only .02% is left.

T. S. 3/4.4.9.1

Justifications:

1. The ACTION statement has been reorganized for clarity to minimize confusion and avoid unnecessary thermal cycling of the Plant.

2. Added plant-specific information regarding specimen location and lead factor. Unit 3 capsule "V" has already been withdrawn.

T. S. 3/4.4.9.2

Justifications:

1. The change reflects the current Plant requirements.

2. The ACTION Statement has been reorganized for clarity to minimize confusion and avoid unnecessary thermal cycling of the Plant.

T. S. 3/4.4.9.3

Justifications:

1. As worded now, the Specifications require that HHSI capability be isolated at exactly 380°F (when cooling down) and implemented at exactly 380°F (when heating up). This is unnecessarily restrictive on the Plant. The revised wording provides the Plant desired flexibility.

T. S. 3/4.4.10

Justifications:

1.

a. Components of the Reactor Coolant System were not designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (1981 edition and addenda through Winter 1981).

b. 1981 Code was not written when Turkey Point Units 3 and 4 were designed.

c. The paragraph order has been revised for clarity.

2.

a. The changes proposed are consistent with applicable code requirements.

b. In addition to the referenced changes, all Class 1, Reactor Coolant System repairs on components shall meet the original construction requirements for that specific repair.

c. Following repair on the RCS pressure boundary, the repaired area receives a hydrostatic presure test in accordance with ASME applicable code requirements which assure the structural integrity of the repaired area.

T. S. 3/4.4.11

Justifications:

1. The surveillance requirement frequency has been changed to refueling invervals because the Plant must be shutdown to perform the surveillance. The 18 month requirement may require an unnecessary shutdown and cycling of the Plant.

T. S. Number 3/4.5.1

Justifications:

1. A limit of 24 months on "refueling" surveillance is a management action to limit time between surveillances.

T. S. 3/4.5.2

Justification:

1. This special report is a duplicate requirement to 10 CFR 50.73 for ESF actuation.

2. As worded now, the tech specs require that HHSI capability be isolated at exactly 380°F (when cooling down) and implemented at exactly 380°F (when heating up). This is unnecessarily restrictive on the Plant. The revised wording provides the Plant desired flexibility.

3. Valve *-887 was removed from the list because it is operated in the fully open position.

   Valve FCV-*-605 was added to the list because it is used as an RHR throttle valve when it is necessary to bypass the heat exchangers or valve HCV-*-758.

4. This comment has been added to provide flexibility required for dual unit shutdown to prevent a severe transient on the Florida electrical grid. This comment is consistent with previous discussions and agreements during RTS working meetings regarding dual unit shutdowns due to electrical components being inoperable.

5. The correct position for valves 866A and B is "closed". FPL drawing 5610-T-E-4510 Sheet 2 shows these valves as normally closed.

6. The surveillance requirement frequency has been changed to refueling intervals because the Plant must be shutdown to perform the surveillance. The 18 month requirement may require an unnecessary shutdown and cycling of the Plant.

T. S. 3/4.5.3

Justification:

1. This special report is a duplicate requirement to 10 CFR 50.73 for ESF actuation.

2. This requirement states that all High Head Safety Injection (HHSI) pump motor breakers shall be racked out when one or more RCS cold legs is less than or equal to 275°F. This is impossible since the system is shared and may be required operable for the other unit whose RCS temperature is greater than 275°F.

T. S. 3/4.5.4

Justification:

1. BASES

This change clarifies that the use of portable instrumentation is acceptable. There is no permanent temperature monitoring instrumentation installed on the RWST.

T. S. <u>3/4.6.1.1</u>

Justifications:

1. We need capability to test certain components, i.e., PASS, where manual containment boundary valves have to be open.

   See recently approved amendments 114/108 dated 6/27/85.

2. Surveillance is not needed. Surveillances 4.6.1.3 ensure compliance with 3.6.1.3 which maintains containment integrity. Also, surveillance is without time frame.

T. S. 3/4.6.1.2

Justifications:

1. L_t is an option which Turkey Point does not intend to utilize. (Also applies to BASES.)

2. ACTION STATEMENT needs to address what must be done if Type B and C leakage exceeds .6 L_a in modes 1, 2, 3 and 4 (RCS temperature greater than 200°F).

   This is consistent with containment integrity Tech Specs.

3. These surveillances are quoting a portion of 10 CFR 50 App J. It has already been clearly established in 4.6.1.2.1 that all types A, B, and C, tests will be performed in accordance with 10 CFR 50 Appendix J. It is confusing to be quoting "portion" of Appendix J requirements. (In fact, Item C.1 as written is in conflict with 10 CFR 50 Appendix J Section III.A.3.b.) Meeting 10 CFR 50 Appendix J <u>in entirety</u> is required by law.

T. S. <u>3/4.6.1.3</u>

Justifications:

1. $.02L_a$ is much less than is expected and has no basis. This number, if implemented, would cause numerous unit shutdowns.

   This paragraph represents a portion of combined leak rate allowable of 0.6 L_a. 0.2 L_a represents a value which, if exceeded, will provide timely indication of air lock seal failure. However, the 0.6 L_a limit of Tech Spec 3.6.1.1 controls the overall containment leakage.

T. S. Number 3/4.6.1.4

Justifications:

1. Revised BASES for consistency with CTS.

T. S. 3/4.6.1.5

Justifications:

1. FPL analysis has determined that containment air temperatures exceeding 120F up to 125F can be accommodated for limited durations.

T. S. 3/4.6.1.6

Justifications:

1. Change is made to maintain consistency on reporting requirements.

2. Based on Turkey Point's CTS, all requirements for adjacent concrete surfaces have been met. The CTS require inspections on the end anchorage concrete surfaces, the mapping of the predominant visible concrete crack patterns, and the measurement of the crack widths. These inspections were done during the Structural Integrity Tests one-half year after the SIT and one year after the SIT. The inspection report determined that the conditions are satisfactory, therefore, the close inspections were terminated.

   Due to the design of the containment at Turkey Point, the inspections of individual tendons and adjacent concrete areas are not easily performed. The equipment and personnel resources expended for this are not justified for the probable limited benefits. Operations to remove grease from tendons and to inspect tendon wire buttonheads for corrosion are already performed during tendon surveillances; inspection of end anchorages and adjacent concrete should be done at the same time, rather than requiring an additional mobilization of personnel and equiment for a second inspection at the time of ILRT. It is not desirable to perform this operation during an outage; the need to position a crane near the containment will interfere with other outage-related activities. Also, note that the same tendons will not be inspected at each surveillance; T.S. 4.6.1.6.1 requires random selection.

   Although the 5-year tendon inspection is greater than the ILRT interval (40 months), the historical data supports this lower inspection frequency. In addition, procedures are in use at Turkey Point which have acceptance criteria based on normal containment pressure rather than ILRT pressure.

3. This visual inspection is already required by 10 CFR 50, App J, Section V.A (which is referenced in Tech Spec 4.6.1.2.1). Both 10 CFR 50 App J and Reg. Guide 1.35, Sections B and C.3, refer to this as a "general" inspection intended to identify "widespread" problem areas. Therefore, no rigid acceptance criteria are provided (as none were recommended in the revised Tech Specs). Additional discussion of acceptable inspection techniques has been added; requirements for record keeping have been added to facilitate the use of data for comparisons to future inspections.

T. S. 3/4.6.1.6 (continued)

4. BASES - Selection of tendons may not be completely random; certain tendons must be eliminated from possible selection because they are inaccesible or would create hazards to personnel performing the surveillance, e.g., tendons located near main steam vents.

5. Proposed Rev. 3 of Regulatory Guide 1.35 recommends the inspection of 2% of each group of tendons (dome, vertical and hoop). For Turkey Point, this equates to 3 dome, 4 vertical and 5 hoop tendons.

6. This change will assure that the appropriate lift-off stress is used for each inspected tendon. Per the FSAR, the average tendon prestress would be less than 0.7 times the ultimate strength; however, different tendons (even within the same group) would have different effective prestress levels after elastic shortening. In addition, time-dependent prestress losses occur (creep, shrinkage and relaxation).

   The FSAR design stress levels are based on the end of the 40-year life of the plant. Comparison of values obtained at an intermediate point in plant life could result in a non-conservative assessment of prestress adequacy.

T. S. <u>3/4.6.1.7</u>

Justifications:

1. The tech spec is reworded to clearly identify the requirements and to reflect previous agreements with NRC as noted below.

   By letter dated November 28, 1978, the NRC requested all licensees to respond to generic concerns about containment purging and venting during normal plant operation. That letter stated that unlimited purging during normal operation would be permitted if licensees demonstrated that purge isolation valves were capable of closing against the dynamic forces of a design basis loss-of-coolant accident. The November 28, 1978 letter also required that licensees evaluate the impact of purging on ECCS performance, the radiological consequences of any design basis accident requiring containment isolation occurring during purge operations and the containment purge and isolation instrumentation and control circuit designs. Pending completion of the above, the NRC requested that licensees commit to cease or limit purging. FPL committed to limit purging for Turkey Point Units 3 and 4 during power operation (2% power) to 200 hours per year for the site (200 hours total for both units). At that time, FPL stated that it intended to justify unlimited purging.

   By letter dated December 13, 1979 (L-79-346), FPL provided the results of the required evaluations. The evaluation of the effect of containment purging on ECCS performance indicated that the effect of purge operation upon the calculated pellet cladding temperatures is small. An assessment of the incremental increase in radiological dose caused by containment purging during a postulated loss-of-coolant accident (LOCA) indated that the anticipated total LOCA dose to be well within the limits of 10 CFR Part 100. An evaluation of the containment purge instrumentation and control design did not identify any single failure concerns.

   The NRC in an evaluation dated August 31, 1981, indicated that the 200 hour purge limit, ECCS analysis, and the instrumentation and control design, were acceptable. In a letter dated February 10, 1983, the NRC provided the results of a generic evaluation of the radiological consequences of accidents while purging or venting at power. To assure that the generic evaluation was valid for Turkey Point, the NRC verified the adequacy of technical specification limits on iodine equilibrium and valve closure times. That evaluation also indicated that the dose contribution through open valves is small, and that the total accident radiological consequences of such

T. S. 3/4.6.1.7 (continued)

such accident would be less than the dose guidelines of 10 CFR Part 100.

FPL letters dated September 17, 1982 (L-82-407), March 4, 1983 (L-83-120), and April 2, 1984 (L-84-86) provided information to demonstrate operability of the purge and vent valves. The NRC safety evaluation report dated August 17, 1984 concluded (subject to replacement of the bolts in the operators and installation of debris screens) that FPL had demonstrated the ability of the valves to close against the buildup of containment pressure in the event of a DBA/LOCA.

The August 17, 1984 NRC letter further stated that FPL's proposed containment purge technical specifications should reflect the limitation of the opening angle for the (48-inch and 54-inch) purge valves, and reflect (in the basis) that the combined purges for both units will be about 200 hours per year during power operation for the site (200 hours total for both units). The proposed technical specifications provided by FPL as part of the technical specification upgrade project were consistent with these requirements. Additional restrictions over and above what the NRC has reviewed and accepted are not justified, and could impose a hardship on plant operations. The guidelines for the upgrade project would preclude the imposition of such requirements.

T. S. 3/4.6.2.2

Justifications:

1. 2000 gpm flow rate cannot be verified during operation unless the CCW system is put in accident configuration (e.g., two RHR Hx in service, all three ECC's operating simultaneously, NCC's isolated, CRDM's isolated, etc.).

   Plant does not contain individual flow indicators for each cooling unit. Valve manipulation required to support this test would result in need to rebalance the CCW system which can only be done during shutdown.

T. S. Number 3/4.6.3

Justifications:

1. This surveillance requires that the plant be in a shutdown condition. An 18 month surveillance frequency may cause FPL to have to shutdown for refueling early or cause an unnecessary outage.

T. S. 3/4.6.4

Justification:

1. This table is derived from the CTS with the exception that the steam generator blowdown valves were deleted from this table. The secondary system is considered to be an extension of containment for containment isolation purposes, and does not rely on any valves, including blowdown valves, to provide containment integrity.

2. Of the Table 3.6-1 isolation valves, only the purge valves are explicitly called out in the LOCA analysis. Specific valve closure times have been eliminated from the tech spec.

3. The removal of Notes 2, 5 and 6 is an editorial change. Note 1 was deleted because this information is not specifically referenced in this tech spec.

   Notes 4 and 7 were deleted due to removal of specific isolation times (see 2 above).

T. S. 3/4.6.6

Justifications:

1. Sample is obtained from the carbon tray after removal. There is no provision to obtain a sample in accordance with NRC Reg. Guide 1-52, Sections C.6.b.

2. This surveillance requires that the plant be in a shutdown condition. An 18 month surveillance frequency may cause FPL to have to shutdown for refueling early or cause an unnecessary outage.

T. S. 3.7.1.1

Justifications:

1. There is no loop isolation capability from S/G's.

T. S. 3/4.7.1.2

Justifications:

1. This comment has been added to provide flexibility required for dual unit shutdown to prevent a severe transient on the Florida electrical grid. This comment is consistent with previous discussions and agreements during RTS working meetings regarding dual unit shutdowns due to electrical components being inoperable.

2. This surveillance requires that the plant be in a shutdown condition. An 18 month surveillance frequency may cause FPL to have to shutdown for refueling early or cause an unnecessary outage.

T. S. 3/4.7.1.3

Justifications:

1. Same ACTION Statement and logic train as Auxiliary Feedwater tech specs.

2. This surveillance requires that the plant be in a shutdown condition. An 18 month surveillance frequency may cause FPL to have to shutdown for refueling early or cause an unnecessary outage.

T. S. 3/4.7.1.4

Justifications:

1. This footnote is confusing. It is FPL's understanding that an internal NRC position has been taken by some technical branches that this footnote should be removed from the STS.

T. S. 3/4.7.1.5

Justifications:

1. This change will allow reasonable time for corrective maintenance and represents a 50% reduction from AOT allowed in CTS.

2. Steam demand can exceed steam produced in MODE 3.

T. S. 3/4.7.1.6

Justifications:

1. Use of the term "operable" has been modified to reflect the fact that these pumps are not safety related. This concept is consistent with CTS (Amendment 118/112, 8/86). Due to plant configuration, this path is normally isolated.

2. This clarification ensures the unit(s) will not be shutdown because of inoperability of non-safety grade equipment. This comment is consistent with CTS (118/112).

3. A limit of 24 months on "refueling" surveillance is a management action to limit time between surveillances.

T. S. Number 3/4.7.2

Justifications:

Note: The CCW Tech Spec was extensively discussed between NRC/FPL in Bethesda on March 15, 1988 (see attached NRC letter to FPL, dated 3/29/88). FPL's comments and justifications reflect the discussion and agreements made at that meeting.

1. The PTN design contains 3 heat exchangers, however, only 2 heat exchangers, capable of removing design basis heat loads, are required. Although provisions are available for isolating passive failures, a passive failure of a heat exchanger is not a postulated design basis. NRC concurred with this approach, in particular noting the severity of the associated ACTION C (plant shutdown - as marked up) and the extensive heat exchanger surveillance program.

   Note that the term "in service" was used and clarified via footnote in lieu of the term "OPERABLE", whose conventional definition may be misinterpretted in this application.

2. "Valves, interlocks, and piping" are considered to be incorporated into the operability requirements for CCW pump/heat exchangers, or the operability of components served by CCW, as appropriate. Accordingly, no specific reference to them is required in the LCO or ACTION.

3. The CCW design incorporated 3-100% capacity pumps. The change of AOT from 24 to 72 hours reflects the remoteness of a scenario which would disable the remaining pump(s) and provides additional time which may be required to effect repairs. Additionally, the requirement to maintain 3 pumps operable is more conservative than current industry practices and the STS.

4. See Justification 1 above. The wording, as marked up, reflects the LCO and system requirements.

5. With this pump inoperable, the plant still retains 2 independent 100% capacity CCW pumps. The CCW pumps have a good maintenance history and a 30 day AOT is reasonable.

6. These surveillances require the plant to be in shutdown condition. An 18 month surveillance frequency may cause FPL to shutdown for refueling early or cause an unnecessary outage.

7. BASES is revised to clarify design requirements for CCW and the reasoning behind the SURVEILLANCE wording.

8. Generic dual shutdown concern.

3/4.7.2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 29, 1988

Docket Nos. 50-250
and 50-251

RECEIVED

APR 05 1988

Nuclear Licensing

LICENSEE: Florida Power and Light Company

FACILITY: Turkey Point Units 3 and 4

SUBJECT: SUMMARY OF MEETING HELD WITH FLORIDA POWER AND LIGHT COMPANY (FP&L) ON MARCH 15, 1988, REGARDING TECHNICAL SPECIFICATIONS AND OPERABILITY OF INTAKE COOLING WATER AND COMPONENT COOLING WATER SYSTEMS

REFERENCE: TAC Numbers 63038 and 63039

A meeting was held in Rockville, Maryland on March 15, 1988 with representatives of Florida Power and Light Company (FP&L) to discuss two matters related to Technical Specifications (TS) for the Intake Cooling Water (ICW) and Component Cooling Water (CCW) systems at Turkey Point Units 3 and 4. The first matter concerned possible interpretations of "operability" of CCW heat exchangers. The second matter related to a revision of the TS for both systems as part of the ongoing TS Revision Project.

1. Operability of CCW Heat Exchangers

In a letter to the licensee (FP&L) dated December 3, 1987, the staff indicated that each CCW heat exchanger should be declared inoperable when it becomes known that it cannot remove its design basis heat load (50% of the required heat removal for the reactor unit). The staff has now improved this interpretation to address the matter from a systems standpoint. The staff agreed with FP&L that it is sufficient if two CCW heat exchangers operating together can remove the total design basis heat load of two CCW heat exchangers, provided that a monitoring program is in place to assure this capability continues to exist. The concern is that fouling of the heat exchanger surfaces could degrade the heat removal capability. To paraphrase, the staff agreed that it is permissible for the plant to operate with one operating CCW heat exchanger which is not able to remove all of its design basis heat load, provided that a second operating CCW heat exchanger can fully compensate by removing more than its design basis heat load, and provided that FP&L monitors the heat exchanger capability on a frequent basis. The licensee agreed to provide a letter to the staff by March 18, 1988 agreeing to these provisions. The staff will then issue a letter clarifying Enclosure 2 of the December 3, 1987 letter.

2. Revision of TS for CCW and ICW

The staff indicated that it could not accept a relaxation of TSs which would entirely remove Allowable Outage Time (AOT) from the third ICW pump and the third CCW pump. This position is based on the current knowledge

of the importance of service water to core melt frequency as identified in the Byron PRA and is under generic review by the staff under Generic Issue 130. The staff suggested that simply relaxing the current 24-hour AOT for the ICW pumps to perhaps 72 hours was a better approach. The staff also suggested that the ICW strainers should be included in the TS with an AOT. The licensee agreed to consider possible AOTs for the ICW and CCW pumps and the ICW strainers and propose these at a later date. In addition, the staff agreed that no AOT is necessary for the third CCW heat exchanger because of the natural incentive for the licensee to maintain the heat exchanger in an operable condition, and because of the TS interpretation described in (1) above.

The meeting agenda and reference material discussed at the meeting are provided as Enclosure 1 to this letter. The attendance list is Enclosure 2.

Gordon E. Edison /

Gordon E. Edison, Sr. Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enclosures:
See next page

T. S. Number 3/4.7.3

Justifications:

Note: The ICW Tech Spec was extensively discussed between NRC/FPL in Bethesda on March 15, 1988 (see attached NRC letter to FPL, dated 3/29/88). Unless otherwise noted below, FPL's comments and justifications reflect the discussions and agreements made at that meeting.

1. "Two headers", as well as "valves, interlocks and piping" are considered to be incorporated into the operability requirements for ICW pumps, or the operability of components served by ICW, as appropriate. Additionally, due to the open design and operation of the system, a specific requirement for headers was deemed unnecessary. Accordingly, no specific reference to them is required.

2. A 3.0.4 exclusion is added, consistent with CCW tech spec and NRC/FPL discussions.

3. The ICW design incorporates 3-100% capacity pumps. The change of AOT from 24 to 72 hours reflects the remoteness of a scenario which would disable the remaining pump(s) and provides additional time which may be required to effect repairs. Additonally, the requirement to maintain 3 pumps operable is more conservative than current industry practices and the STS.

4. These surveillances require the plant to be in a shutdown condition. An 18 month surveillance frequency may cause FPL to shutdown for refueling early or cause an unnecessary outage.

5. BASES is revised to clarify that design "and operation" of this system ensures required cooling capacity. This clarification recognizes that operator actions may be required in certain scenarios, as previously discussed with NRC.

6. As discussed between FPL/NRC at the March 15th meeting, FPL is to propose wording regarding the basket strainers. This wording will be provided at a later date.

3/4.7.3

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

March 29, 1988

Docket Nos. 50-250
and 50-251

RECEIVED
APR 05 1988
Nuclear Licensing

LICENSEE: Florida Power and Light Company

FACILITY: Turkey Point Units 3 and 4

SUBJECT: SUMMARY OF MEETING HELD WITH FLORIDA POWER AND LIGHT COMPANY (FP&L) ON MARCH 15, 1988, REGARDING TECHNICAL SPECIFICATIONS AND OPERABILITY OF INTAKE COOLING WATER AND COMPONENT COOLING WATER SYSTEMS

REFERENCE: TAC Numbers 63038 and 63039

A meeting was held in Rockville, Maryland on March 15, 1988 with representatives of Florida Power and Light Company (FP&L) to discuss two matters related to Technical Specifications (TS) for the Intake Cooling Water (ICW) and Component Cooling Water (CCW) systems at Turkey Point Units 3 and 4. The first matter concerned possible interpretations of "operability" of CCW heat exchangers. The second matter related to a revision of the TS for both systems as part of the ongoing TS Revision Project.

1. Operability of CCW Heat Exchangers

In a letter to the licensee (FP&L) dated December 3, 1987, the staff indicated that each CCW heat exchanger should be declared inoperable when it becomes known that it cannot remove its design basis heat load (50% of the required heat removal for the reactor unit). The staff has now improved this interpretation to address the matter from a systems standpoint. The staff agreed with FP&L that it is sufficient if two CCW heat exchangers operating together can remove the total design basis heat load of two CCW heat exchangers, provided that a monitoring program is in place to assure this capability continues to exist. The concern is that fouling of the heat exchanger surfaces could degrade the heat removal capability. To paraphrase, the staff agreed that it is permissible for the plant to operate with one operating CCW heat exchanger which is not able to remove all of its design basis heat load, provided that a second operating CCW heat exchanger can fully compensate by removing more than its design basis heat load, and provided that FP&L monitors the heat exchanger capability on a frequent basis. The licensee agreed to provide a letter to the staff by March 18, 1988 agreeing to these provisions. The staff will then issue a letter clarifying Enclosure 2 of the December 3, 1987 letter.

2. Revision of TS for CCW and ICW

The staff indicated that it could not accept a relaxation of TSs which would entirely remove Allowable Outage Time (AOT) from the third ICW pump and the third CCW pump. This position is based on the current knowledge

of the importance of service water to core melt frequency as identified in the Byron PRA and is under generic review by the staff under Generic Issue 130. The staff suggested that simply relaxing the current 24-hour AOT for the ICW pumps to perhaps 72 hours was a better approach. The staff also suggested that the ICW strainers should be included in the TS with an AOT. The licensee agreed to consider possible AOTs for the ICW and CCW pumps and the ICW strainers and propose these at a later date. In addition, the staff agreed that no AOT is necessary for the third CCW heat exchanger because of the natural incentive for the licensee to maintain the heat exchanger in an operable condition, and because of the TS interpretation described in (1) above.

The meeting agenda and reference material discussed at the meeting are provided as Enclosure 1 to this letter. The attendance list is Enclosure 2.

Gordon E. Edison /

Gordon E. Edison, Sr. Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures: As stated

cc w/enclosures:
See next page

T. S. 3/4.7.4

Justifications:

1. The requirements of this tech spec are incorporated into the CCW tech spec through the requirement to monitor and maintain in service CCW heat exchangers capable of removing design basis heat loads. A tech spec for a specific ultimate heat sink temperature is inappropriate; ultimate heat sink temperature is one of several factors (i.e., heat exchanger performance characteristics, flow) involved in determining heat removal capability.

T. S. 3/4.7.5

Justifications:

1. As there is only one (1) CREVS, the word "the" was substituted for "each" for clarity.

2. The added words had been deleted by the NRC. Their deletion would imply the requirement to visually inspect the entire CREVS. This would require a major effort, as large parts of the system are not accessible without disassembly.

3. CTS AOT is 3-1/2 days. STS is seven (7) days for a redundant system (e.g., two (2) independent systems).

4. This comment has been added to provide flexibility required for dual unit shutdown to prevent a severe transient on the Florida electrical grid. This comment is consistent with previous discussions and agreements during RTS working meetings regarding dual unit shutdowns due to electrical components being inoperable.

T. S. 3/4.7.6

Justifications:

1. This inspection cannot be done at power.
2. Footnote clarifies "inaccessible".
3. A limit of 24 months on "refueling" surveillances is a management action to limit time between surveillances.

T. S. 3/4.7.8.2

Justifications:

1. Per the Turkey Point Appendix R upgrade, the listed spray/sprinkler systems were superseded for the following reasons:

   a. 4160V Switchgear Room Louver Spray - This system was rendered obsolete when the opening it was protecting was filled with a 3-hour fire barrier.

   b. EDG Building Water Curtain - This manual system on the outside of the EDG Building was superseded by an automatic suppression system in the EDG rooms.

   c. Control Room Guardhouse Sprinkler System - This system protected the old guardhouse, which was constructed of combustible materials. A non-combustible guardhouse has since been constructed.

T. S. 3/4.7.8.3

Justifications:

1. These hose stations are not all shared.

T. S. Number 3/4.7.9

Justifications:

1. BASES - This evaluation is inconsistent with the requirements of the LCO.

T. S. 3/4.9.1

Justification:

1. RTS wording is revised to 0.95 per Amendment 132/126. The restriction on K_{eff} is CTS and has been lessened by this recent amendment. (This justification also applies to BASES.)

2. The term "equivalent" would mean 100^+ gpm of 1950 ppm; this would require 2 charging pumps in Mode 6. No rate requirements for boration in Mode 6.

3. The tech spec is not requiring valves be "locked closed".

T. S. Number <u>3/4.9.8.1</u>

Justifications:

1. Due to expected maintenance activities during this mode, the suggested 4 hour time period may not be achievable.

2. A limit of 24 months on "refueling" surveillances is a management action to limit time between surveillances.

T. S. Number 3/4.9.8.2

Justifications:

1. Due to expected maintenance activities during this mode, the suggested 4 hour time period may not be achievable.

T. S. Number 3/4.9.9

Justifications:

1. Consistent with tech spec 3.3.2 (Table 3.3-2, item 3.c.4) for ESFAS operability in Modes 1-4.

T. S. Number 3/4.9.11

Justifications:

1. There is no need for an upper limit on water level.

T. S. 3/4.9.12

Justification:

1. All design calculations are based on the time the fuel was last critical, not the time the fuel was placed in the Spent Fuel Pool.

2. Wording was clarified to state 120 days of decay . . . not 120 days in cask.

T. S. Number 3/4.9.13

Justifications:

1. Consistent with Tech Spec 3.3.2 (Table 3.3-2, item 3.c.4) for ESFAS operability in Modes 1-4.

T. S. 3/4.9.14

Justification:

1. BASES - Current Technical Specification 3.17 specifies a maximum K_{eff} of 0.95 applicable to the proposed two-region spent fuel pools at PTN. This is also discussed in RTS 5.6.1.

   Refer to Justification 3/4.9.1, #1.

T. S. Number 3/4.10.1

Justifications:

1. The term "equivalent" would mean 100+ gpm of 1950 ppm; this would require two (2) charging pumps. Tech Spec 3.1.2.3 ACTION allows two (2) charging pumps to be inoperable for a limited time.

T. S. 3/4.10.3

Justification:

1. Depending on specific unit configuration, the operating staff needs flexibility to deal with the LCO without potentially creating an unexpected operating condition.

T. S. Number 3/4.10.5

Justifications:

1. Recently approved evaluation by FPL's Fuels Analysis Group allowed withdrawal of two banks during control rod testing.

T. S. Number 3/4.11.2.1

Justifications:

1. The Unit 3 and 4 containment bleed lines constitute a small release rate to the plant vent. This bleed rate is continuous unless containment isolation occurs, at which time the bleed lines are automatically isolated. All realses via this pathway are continuously monitored by the installed plant vent monitors for noble gas, particulate and iodine activities. This monitoring will account for all activity released from the containment by the bleed line pathway.

2. Note 6: No need to sample if coolant activity has gone up unless the effluent monitor has also gone up. Therefore, the statement should require both conditions. (This wording is consistent with recently issued specifications at St. Lucie and Palo Verde.)

3. Note 7: FPL's comment returns the wording to a positive format where analysis is required when both requirements are met; with the negative format, it appears analysis is required when either one or both of the conditions are not met. Without this comment, this tech spec would represent a significant change to the CTS (Amendment 103/97).

T. S. Number 3/4.11.2.5

Justifications:

1. The plant has the ability to monitor only the in-service tank, not all 6 tanks simultaneously. This comment is consistent with previous FPL/NRC discussions.

2. Footnote has been added for consistency with Table 3.3-8, Item 2 which covers the use of grab samples for inoperability of continuous monitors.

T. S. Number 3/4.11.3

Justifications:

1. All of our shipments do not go directly to a burial site, some are shipped to a waste processor for volume reduction prior to burial.

   Only "wet" radioactive wastes (not all radioactive wastes) are required to be dewatered or solidified.

T. S. Number <u>3/4.11.4</u>

Justifications:

1. BASES - Turkey Point's outside storage tanks, if released, will be contained within the site boundary and mixed with our cooling canals which are completely enclosed.

T. S. Number 3/4.12.1

Justifications:

1. The existing requirement (if sample is >1.0 units and >10 times most recent control sample) is considered appropriate to trigger gamma isotopic analysis of specific samples. The monitoring program is conducted by the State of Florida under a joint Turkey Point/St. Lucie contract. This comment provides a consistent approach between Turkey Point and St. Lucie and reflects the requirements of the CTS (RETS Amendment 103/97).

T. S. Number 3/4.12.3

Justifications:

1. The State Laboratory participates in the EPA Intercomparison Program for media, isotopes and analysis not required by our program in addition to those that are required. The proposed wording would have FPL generate a report if the State chose not to participate in one or more of the "other" media, isotopes or analysis.

T. S. Number _5.1, Figure 5.1-1_

Justifications:

1. This figure will be redrawn to reflect the required level of detail. An original of the redrawn figure will be provided to the NRC.

T. S. Number 5.3

Justifications:

1. Initial fuel load enrichment is historical information and is not required. Reload fuel enrichment will be a changing value and will be adequately documented in the reload analyses provided in the FSAR.

2. This wording reflects recent CTS Amendment.

T. S. Number 5.7, Table 5.7-1

Justifications:

1. These changes are the result of an engineering evaluation of cyclic loading. An FSAR change is being processed which will change these values in FSAR Table 4.1-8.

T. S. Number 6.2

Justifications:

1. The Operations Superintendent is a senior manager with responsibility for operations, outage scheduling, chemistry, reactor engineering, and health physics. In the absence of the Plant Manager, the Operations Superintendent normally assumes the duties of the Plant Manager. The Turkey Point Plant organization does not have an Assistant Plant Manager position. The proposed requirement that the Operations Superintendent either hold or have held an SRO license would continue to ensure that he has the knowledge and experience commensurate with his level of responsibility. The Operations Supervisor, who has direct responsibility for operations, will continue to hold a current SRO license. These requirements provide reasonable assurance that decisions and actions during normal and abnormal conditions will be such that the plant will be operated in a safe and efficient manner.

T. S. Number 6.3

Justifications:

1. This change is required for consistency with the proposed change to 6.2.2.i.

2. The March 28, 1980 NRC letter has been superseded. RO and SRO qualifications are now covered by 10 CFR 55 and ANSI 3.1, 1981.

T. S. Number 6.4

Justifications:

1. The 10 CFR 55 requirements have been included in the body of the document, and are no longer in Appendix A. The March 28, 1980 NRC letter has been superseded by 10 CFR 55 and ANSI 3.1, 1981.

T. S. Number 6.5

Justifications:

1. These changes were made to reduce management time spent on PNSC activities. The problem of excessive PNSC meetings and line management involvement was noted by the NRC in Safety Review Team Inspection Report 87-49 (attached). In response to this concern, FPL proposes adapting the Palo Verde Unit 3 T.S. regarding technical review and control activities. This T.S. removes the direct responsibility for review of various plant documents and programs from the PNSC. FPL believes that the delegation of review responsibility adequately responds to the NRC's concern, and is consistent with the NRC's interpretation of PNSC responsibilities (various memos attached).

2. This type of generic composition of the CNRB has been previously approved by the NRC for other plants (e.g., Fermi II). It will eliminate the need for future license amendments due to changes in titles of CNRB members.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

T.S. 6.5

February 10, 1988

RECEIVED
FEB 12 1988
Nuclear Licensing

Docket Nos. 50-250
and 50-251

Florida Power and Light Company
ATTN: Mr. C. O. Woody
Executive Vice President
P. O. Box 14000
Juno Beach, Florida 33408

SUBJECT: SAFETY REVIEW TEAM INSPECTION REPORT NO. 50-250/87-49; 50-251/87-49

Gentlemen:

This letter forwards the report of the Safety Review Inspection conducted by Mr. C. Haughney and other NRC personnel during the period December 7-11, 1987, of activities at the Turkey Point Power Plant, authorized by NRC Operating Licenses DPR-31 and DPR-41, and to the discussion of our findings with Mr. C. Baker and others at the conclusion of the inspection.

The inspection consisted of an examination, on a sampling basis, of activities in the areas of safety review pursuant to 10 CFR 50.59, and the on-site and off-site review committees. The inspection included assessment of safety review documentation, interviews with site and corporate personnel, and attendance at committee meetings.

The inspection team found evidence of significant improvement in the quality of safety evaluations over the past year. Remaining steps to assure continuation of this trend include the timely issuance of your comprehensive draft procedure covering 10 CFR 50.59 evaluations and associated training for reviewers.

Additional areas for improvement identified by the inspection team include: improving the efficiency of Plant Nuclear Safety Committee (PNSC) reviews, identifying currently valid safety evaluations, classification of guidance documents regarding active check valve failures, the retention of design calculations and timely completion of Request for Assistance evaluations (REAs).

Some of the items identified by the team may be potential enforcement findings. Any enforcement actions will be identified by Region II.

In accordance with 10 CFR 2.790(a), a copy of this letter and the enclosure will be placed in the NRC Public Document Room.

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Should you have any questions concerning this inspection, please contact me or Mr. L. Norrholm (301-492-0956) of this office.

Steven A. Varga, Director
Division of Reactor Projects, I/II
Office of Nuclear Reactor Regulation

Enclosure: Inspection Report 50-250/87-49; 50-251/87-49

cc w/enclosure: See next page

T.S. 6.5

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

DIVISION OF REACTOR INSPECTION AND SAFEGUARDS

Report No.: 50-250/87-49 and 50-251/87-49

Docket No.: 50-250 and 50-251

Licensee: Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408

Inspection At: Turkey Point Power Plant
Florida City, Florida

Inspection Conducted: December 7-11, 1987

Team Leader: Charles J. Haughney 1/21/88
Charles J. Haughney, Chief — Date Signed
Special Inspection Branch, DRIS, NRR

Team Members: Leif J. Norrholm 1/20/88
Leif J. Norrholm, Chief — Date Signed
Team Inspection Appraisal and Development Sec. 1

James E. Cummins 1/20/88
James E. Cummins, SRI, Wolf Creek — Date Signed

Consultant: Gary J. Overbeck

Approved By: Charles J. Haughney 1/21/88
Charles J. Haughney, Chief — Date Signed
Special Inspection Branch, DRIS, NRR

T.S. 6.5

1.0 Inspection Scope

This one-week team inspection evaluated safety review activities conducted by the licensee pursuant to Title 10 Code of Federal Regulations Part 50.59. Related functions of the Plant Nuclear Safety Committee (PNSC) and the Corporate Nuclear Review Board (CNRB) were also assessed. The inspection consisted of record reviews, interviews with cognizant personnel, and attendance at meetings. The purpose of the inspection was to determine whether appropriate issues were subjected to 10 CFR 50.59 reviews, whether correct determinations were made with respect to unreviewed safety questions, and whether documentation of these reviews were complete in describing the bases and rationale for conclusions.

2.0 Summary of Significant Findings

Steady improvement in the quality and completeness of 10 CFR 50.59 safety evaluation documentation was observed over the past year. Further improvement and the continuation of current skills in this area should be aided by licensee initiatives to issue comprehensive procedural guidelines and to provide training to evaluators.

In general, recent safety evaluations reviewed were sufficiently detailed to demonstrate, as a stand-alone document, the logic and bases for determinations regarding potential unreviewed safety questions. One notable strength was the use of prior 10 CFR 50.59 review and Plant Nuclear Safety Committee (PNSC) authorization for temporary modifications to the plant, including jumpers and lifted leads.

Some areas were identified as weaknesses which should receive attention by the licensee. Despite attempts at mitigation, the volume of material requiring PNSC review has resulted in long and frequent meetings, some brief reviews, and diversion of management from normal duties. Some safety evaluations reviewed were no longer valid, but no system was in place to identify these. Licensee attention should be directed to insuring that Request for Engineering Assistance (REA) evaluations were completed in a timely manner. Finally, the licensee should reevaluate its position regarding the treatment of check valve active failures and the retention of design calculations.

3.0 Design Changes

3.1 Design Change Process

Licensee procedures for all design and safety analyses performed by the Power Plant Engineering Department were contained in Quality Instruction JPE-QI-3.2, Revision 3, dated July 30, 1982. The guidance provided in this procedure was minimal with respect to 10 CFR 50.59 evaluations in that the definition of unreviewed safety question was quoted from the rule, and an additional statement was made to the effect that the discussion should clearly indicate the bases for the conclusions reached.

Station Administrative Procedure 0190.15, Plant Changes and Modifications (PC/M), dated October 26, 1987, provided essentially the same information as JPE-QI-3.2. Station Administrative Procedure 0190.22, Changes, Tests, and Experiments, dated December 12, 1986, offered some additional guidance

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5.2 Plant Nuclear Safety Committee (PNSC)

Inspection of PNSC consisted of discussions with licensee personnel, including PNSC members, review of documents, and attendance at the PNSC meeting held on December 8, 1987.

One of the events that prompted this inspection was PNSC Meeting No. 86-232 held on August 31, 1986. That meeting was conducted by means of individual telephone calls from the Shift Technical Advisor to each of the PNSC members. PNSC Meeting 86-232 was conducted to get concurrence from the PNSC to start up Unit 4 without first repairing a small, identified reactor coolant leak. The leak was through an instrumentation port column assembly conoseal fitting located on the reactor vessel head. The plant was subsequently restarted and operated at power. During an outage in March 1987, the licensee determined that a significant amount of boric acid from the leak had accumulated on the reactor vessel head. At that time, the licensee also determined that corrosion rates of materials, in contact with the boric acid, may have been greater than those on which the PNSC based the decision to startup the plant in August 1986. This event was documented in licensee Letter No. L-87-186, dated April 27, 1987, to the U.S. Nuclear Regulatory Commission. The consequences of this event indicated that the practice of obtaining PNSC concurrence by walking-around items or by serial telephone calls can result in less than adequate reviews.

The licensee revised Administrative Procedure (AP) 0110.4, "Plant Nuclear Safety Committee General Procedure," dated October 6, 1987, to clarify and tighten the requirements for holding PNSC meetings. The October 6, 1987 revision of AP 0110.4, did not allow walk-around PNSC concurrences. The procedure also required that telcon PNSC meetings were to be conference call type, with the members talking to each other. The following items could not be approved utilizing the telcon PNSC meeting:

- Any item which required a written safety evaluation for approval;
- Any item which involved a change in the FSAR or Technical Specifications;
- Any plant changes or modifications, controlled plant work orders, and process sheets.

One of the licensee personnel interviewed by the team was the Chairman of the PNSC, the Plant Manager-Nuclear. The Plant Manager-Nuclear did not chair many of the PNSC meetings; instead, the majority of the PNSC meetings were chaired by the vice chairman, the Operations Superintendent-Nuclear. The plant manager stated that he intentionally did not chair many of the meetings because by having the vice chairman conduct the meetings, a more independent review of PNSC items was obtained. He felt this practice was necessary because, as plant manager, he had to review each item and give final approval for it to be issued.

The team verified by review of licensee quality assurance documents and discussions with licensee quality assurance personnel, that the licensee's quality assurance department was monitoring PNSC activities. Two of the QA documents reviewed were QA audits QAO-PTN-86-723 and QAO-PTN-87-818. These annual audits were conducted to verify that the PNSC was meeting the requirements of TS 6.5.1 and the related plant implementing procedures. A third QA document reviewed was Corrective Action Request (CAR), Unit 4 Conoseal Leak, CAR-87-019. This

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CAR reported noncompliances identified in a QA performance monitoring activity of events related to the Unit 4 conoseal leak. CAR-87-019 reported that PNSC Meeting 86-232 was conducted on August 31, 1986, by the shift technical advisor making individual telephone calls to each PNSC member, and that it was common practice to obtain PNSC concurrences in this way rather than by conference calls, as now specified by AP 0110.4.

The team made the following two observations during the inspection:

1. The licensee did not have a formal training program for PNSC members or their alternates. In 1984, the licensee held a one-day training session, which was taught by a contractor, for the PNSC members. In June 1987, the PNSC coordinator prepared a PNSC training guideline manual which contained material relative to the operation of the PNSC. This manual was a required reading type training course. These training manuals were sent to each of the PNSC members with a letter stating that the manual was to provide training for the PNSC members and their alternates. At the time of this NRC inspection, the PNSC coordinator had received documentation back from four of the members showing that they and their alternates had reviewed the material referenced in the training manual. At the exit meeting, the licensee stated that they were developing training programs for PNSC members and their alternates.

2. The second observation has to do with the large volume of material that is reviewed by the PNSC. The PNSC was meeting a minimum of twice a week with each meeting lasting over two hours. In addition to the two scheduled meetings every week, call meetings were frequently held. Through December 13, 1987, 334 PNSC meetings had been held during 1987. Discussions with licensee personnel indicated that the minimum of twice-weekly meetings had been held for a number of years. Because of the volume of material and the frequency of the meetings, the meeting agenda and package of items to be reviewed were not provided to the members until they arrived at the meeting. This practice did not give the members an opportunity to familiarize themselves with the items on the agenda. To help alleviate the problem of the large number of items to be reviewed, the PNSC has required that a sponsor be present at the meeting to present each item. The sponsor provided expertise to the PNSC on each item presented. Any item that did not have a sponsor present at the meeting was tabled. In addition, any item that any member questioned was sent back to the preparer to resolve these questions prior to PNSC approval.

This observation describes a condition that is common to many facilities with the traditional standard TS for their onsite review committee. This TS requirement, which applies to Turkey Point, requires that the PNSC review the types of procedures contained in Appendix A to Regulatory Guide 1.33. This appendix lists most of the types of procedures contained in a nuclear plant site's procedure file, and, as a result, the number of procedures subject to review, can be several thousands. This large review task is further compounded by the fact that many of these procedures could undergo changes several times a year. Interviews with PNSC members revealed the not surprising statistic that the vast majority of PNSC meeting time was devoted to procedure and procedure change review.

T.S. 6.5

In addition to the requirement to review procedures and procedure changes, TS 6.5.1.6.f requires a review of facility operations to detect potential safety hazards. When PNSC members were asked how they conducted such a review they were initially unable to respond. After some thought, one member suggested that such a review was accomplished by the review of Licensee Event Reports (LERs). However, the team pointed out that review of LERs was more explicitly covered by TS 6.1.5.6.k, which required review of all reportable events. The team also asked PNSC members whether they had ever been requested to perform special reviews and investigations by the CNRB. These reviews are a TS requirement in 6.5.1.6.g. None of the PNSC members interviewed could recall conducting such a review. Admittedly, both of these TS requirements are very broad and the absence of their reviews by the PNSC would not, in and of itself, constitute a violation of TS. However, the team questioned PNSC members as to whether they had ever made those review requirements meeting agenda items. In the team's view, the PNSC is so burdened with the extensive review of routine procedure changes that it is effectively unable to devote time to reflect upon broad safety issues. This issue is compounded by the fact that PNSC members are key station managers who must divide their time among PNSC activities, plant tours, and supervision of their departments. The team considers that these key people could spend their time more effectively if they could develop alternative means for independent review of many of the station procedures. Certain types of key procedures, such as emergency operating procedures and administrative procedures, could be left for full PNSC review, while other procedures could be reviewed by another process. During the exit meeting the licensee indicated that they would consider requesting a TS amendment and instituting appropriate administrative control changes that would allow them to reduce the procedure review workload on the PNSC.

Based on this limited inspection, the team determined the following:

- The PNSC was conducting adequate reviews.
- Adequate administrative procedures had been implemented to control PNSC activities so that TS requirements were being met.
- The QA department had performed monitoring activities and audits of PNSC activities.

6.0 Exit Meeting

The inspection team conducted an exit meeting on December 11, 1987, to provide a summary of issues identified during the inspection. The licensee's representatives at the exit meeting are identified in Attachment A. The scope of the inspection was discussed, the observations were presented for each area inspected, and team members responded to questions from the licensee representatives.

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McGough
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 16, 1978

Docket No. 50-313

Arkansas Power & Light Company
ATTN: Mr. William Cavanaugh, III
Executive Director, Generation
and Construction
Post Office Box 551
Little Rock, Arkansas 72203

Gentlemen:

By letter dated January 10, 1978, you requested an interpretation of Technical Specification 6.5.1.6 for Arkansas Nuclear One - Unit No. 1. This request was the result of a difference in interpretation between Arkansas Power & Light Company staff and NRC Regional Inspection and Enforcement personnel, and concerns the technical specification wording which states that: "The Plant Safety Committee (PSC) shall be responsible for review of ...". Your interpretation of this requirement is that items of review for which the PSC is responsible could be delegated to individuals or groups outside the PSC. You state that this interpretation is contrary to that of OI&E personnel who require that the PSC perform all reviews and investigations.

The proper interpretation of this provision is as follows:

> The PSC may delegate review responsibility to individuals or groups outside of the formal composition of the PSC. However, since the PSC is responsible for the performance of these reviews, the review function is not complete until the PSC acts upon the results of the review in a formal manner and documents such action. Documentation may take the form of a simple notation in the minutes of a PSC meeting or may be as elaborate as the preparation of a formal PSC report.

T.S. 6.5 J. McDoug

Section 6.0

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 28, 1978

MEMORANDUM FOR: B. H. Grier, Director, RI
J. P. O'Reilly, Director, RII
J. G. Keppler, Director, RIII
G. L. Madsen, Acting Director, RIV
R. H. Engelken, Director, RV

FROM: J. H. Sniezek, A/D for Field Coordination, DROI, IE

SUBJECT: INTERPRETATION OF TECHNICAL SPECIFICATION REQUIREMENTS FOR ON-SITE COMMITTEE REVIEW FUNCTION

On August 12, 1977, I forwarded a memorandum to all RO&NS Branch Chiefs regarding the on-site committee review function. Included in this memorandum was an interpretation by K. R. Goller that was intended to clarify how the on-site committee was intended to function. By memorandum to Karl Goller dated August 2, 1977, Jim Murray said that he had reviewed Goller's memorandum of June 27 and could not agree with the legal conclusions and that he would be unable to support enforcement action in those instances where the On-Site Review Committee did not formally convene a quorum to conduct committee business. Since that time we have held several meetings with NRR and ELD. There appears to be no conflict regarding the intent of committee review requirements. NRR expects the on-site review committee to act on tasks in their area of responsibility while the committee is in session and to record the results of their actions in committee minutes. This is not to say that the individuals must meet to review issues, only they must meet to act on the various issues and that these actions should be recorded. Certain portions of the on-site committee's overall responsibility may be reviewed by subcommittees or individuals who make recommendations to the on-site committee. The on-site committee must act on these recommendations; however, it is not required that the on-site committee actually conduct a detailed review of the issue in question. DROI and ELD concur with NRR's views on this matter.

With regard to enforceability of current technical specification requirements in this area, we note that the above interpretation of how the committee should function is completely consistent with NRC's long standing interpretation of the on-site review function. While we feel that this intent is implicit in the language of the TS, we cannot disagree with ELD's view that the language of the TS may not be

enforceable should the matter be taken to a hearing. To resolve this issue for future plants, NRR intends to revise the language of the STS. With regard to those facilities that are already licensed, it was pointed out that the intent of this specification appears to be clearly understood by all but a few licensees. Therefore, to avoid unnecessary TS changes NRR will request licensees who are not meeting the intent of the specifications to submit a change to their TS. In consideration of the difficulties of a generic resolution to this issue, we find this approach acceptable.

Please call if you have any questions on this matter.

J H Sniezek

J. H. Sniezek, A/D for Field Coordination
Division of Reactor Operations Inspection, IE

cc: RO&NS BCs, Regions I-V
Enforcement Coordinators, Regions I-V
E. L. Jordan
J. Murray, ELD
J. McGough
T. J. Carter

CONTACT: G. L. Constable
(492-8019)

T. S. Number 6.8

Justifications:

1. This change reflects the current T.S. wording.

2. The footnote was deleted because FPL considers it inappropriate to include state regulations in the T.S. This footnote is not in the current T.S.

3. Current T.S. 6.14 contains the same requirements as RTS 6.8.4.a; however, no specific systems are identified. FPL's position is that inclusion of systems could be interpreted as limiting the program to those listed systems.

4. This change reflects the changes to 6.5.1 and 6.5.2.

T. S. Number 6.9

Justifications:

1. The initial startup reports were completed in 1972 and 1973, and met the requirements of the original FSAR. The FSAR has subsequently been revised several times. It would not be productive to compare the initial Startup Reports to the current FSAR. All subsequent Startup Reports will meet or exceed the current FSAR. The proposed wording is taken from the current T.S.

2. This wording is in the current T.S., and reflects FPL's NRC-approved response to NUREG-0472.

3. There are no other uranium fuel cycle sources near the Turkey Point site.

4. This change, which replaces the P&R version with current T.S.6.9.3.d (except for reporting requirements), reflects the FPL-proposed change to T.S. 3.2.2. The NRC-proposed T.S. would require FPL to obtain additional analysis to generate the proposed Peaking Factor Limit Report. Current T.S. 6.9.3.d was approved by the NRC on March 17, 1982.

5. These changes were made to conform with 49 CFR.

6. This is a duplication of the requirements of the Annual Environmental Monitoring Report, and is not part of STS.

7. This change is for consistency with other reporting requirements of 10 CFR 50.4 and T.S.6.9.1.

T. S. Number 6.10

Justifications:

1. These current T.S. words were recently approved (Amendment 103/97), and are consistent with St. Lucie Units 1 & 2. Any change to these requirements would result in program inconsistenciés between FPL's nuclear sites.

T. S. Number 6.13

Justifications:

1. The current T.S. changed per NUREG-0472, contain the requirement to have the PCP reviewed by the PNSC prior to implementation.

T. S. Number 6.15

Justifications:

1. The NRC safety evaluation for License Amendment 103/97 stated that a specification addressing major changes to the radwaste treatment systems was not required. Also, the intent of NUREG-0472 requirements are met since major changes would be reported in the annual FSAR update.