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Technical Specifications
Virgil C. Summer Nuclear Station, Unit No. 1

Docket No. 50-395

Appendix "A" to
License No. NPF-12

**Issued by the
U.S. Nuclear Regulatory
Commission**

Office of Nuclear Reactor Regulation

August 1982

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SECTION 1.0

DEFINITIONS

1.0 DEFINITIONS

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

ACTION

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

ACTUATION LOGIC TEST

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

ANALOG CHANNEL OPERATIONAL TEST

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

AXIAL FLUX DIFFERENCE

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

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

b. All equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

DOSE EQUIVALENT I-131

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

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 for valves that are open under administrative control as permitted by Specification 3.6.4,

b. All equipment hatches are closed and sealed,

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

d. The containment leakage rates are within the limits of Specification 6.8.4.g, and

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

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

CORE OPERATING LIMITS REPORT

1.9a The CORE OPERATING LIMITS REPORT (COLR) is the unit specific document that provides core operating limits for the current operating reload cycle. The cycle specific core operating limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these operating limits is addressed in individual specifications.

DOSE EQUIVALENT 1-131

1.10 DOSE EQUIVALENT 1-131 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 ICRP 30, Supplement to Part I, pages 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.12 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

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

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

MASTER RELAY TEST

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.17 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from 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. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.6 and 6.9.1.8.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

1.22 The PROCESS CONTROL PROGRAM (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 wil be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

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

QUADRANT POWER TILT RATIO

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

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2900 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until loss of stationary gripper coil voltage. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and the methodology for verification have been previously reviewed and approved by the NRC.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

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

SLAVE RELAY TEST

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

1.30 Not Used

SOURCE CHECK

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

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

TRIP ACTUATING DEVICE OPERATIONAL TEST

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

UNIDENTIFIED LEAKAGE

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

TABLE 1.1

OPERATIONAL MODES

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

* Excluding decay heat.

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

TABLE 1.2

FREQUENCY NOTATION

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

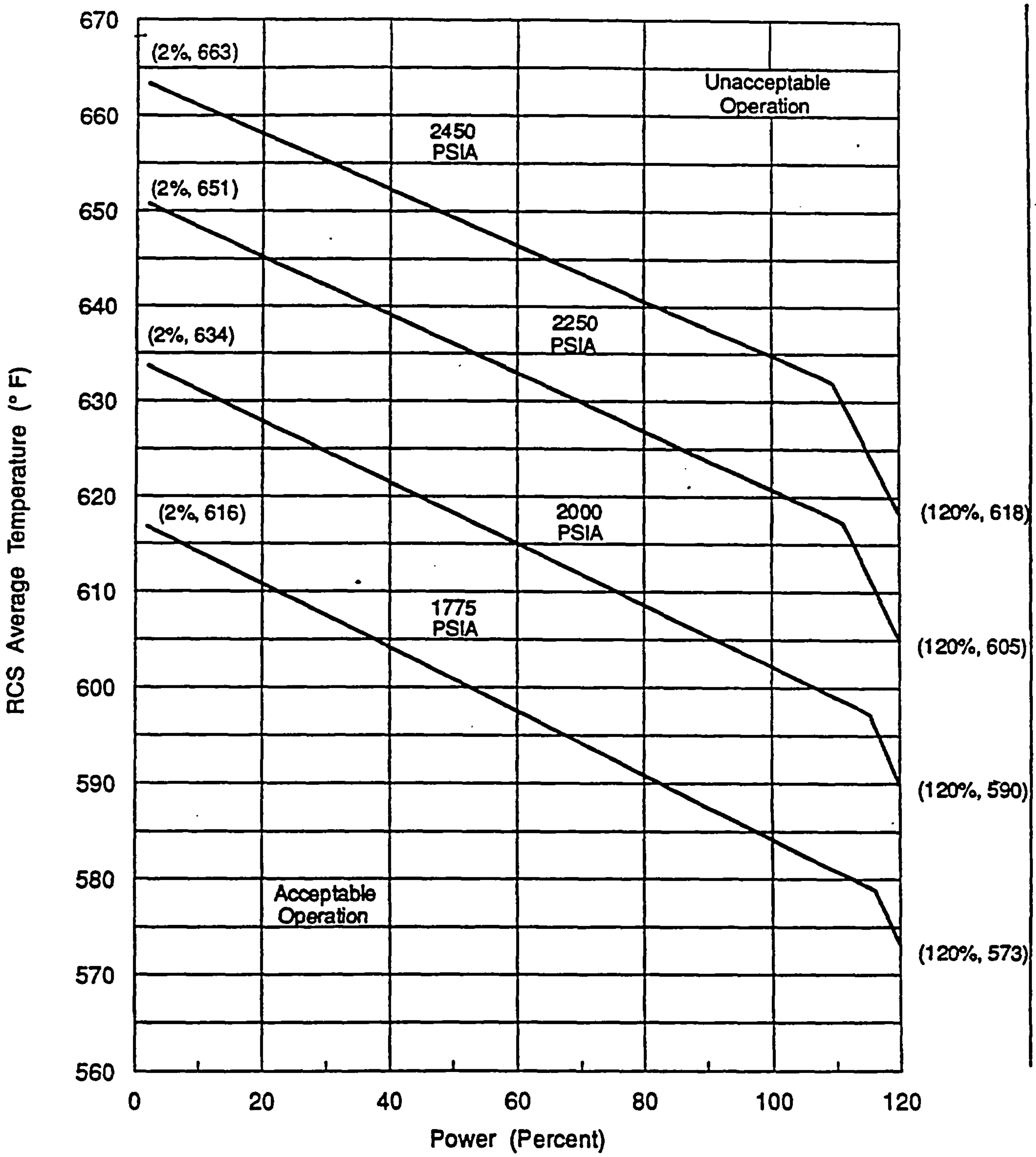
ACTION:

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.



When operating in the reduced RTP region of Technical Specification 3.2.3 the restricted power level must be considered 100% RTP for this figure.

Figure 2.1-1
Reactor Core Safety Limits - Three Loop Operation

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

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

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

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

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

where:

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

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

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

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

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

ACTION:

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

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

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
1.	Manual Reactor Trip	NA	NA
2.	Power Range, Neutron Flux High Setpoint Low Setpoint	 $\leq$109% of RTP $\leq$25% of RTP	 $\leq$111.2% of RTP $\leq$27.2% of RTP
3.	Power Range, Neutron Flux High Positive Rate	$\leq$5% of RTP with a time constant $\geq$2 seconds	$\leq$6.3% of RTP with a time constant $\geq$2 seconds
4.	DELETED		
5.	Intermediate Range, Neutron Flux	$\leq$25% of RTP	$\leq$31% of RTP
6.	Source Range, Neutron Flux	$\leq 10^5$ cps	$\leq 1.4 \times 10^5$ cps
7.	Overtemperature ΔT	See note 1	See note 2
8.	Overpower ΔT	See note 3	See note 4
9.	Pressurizer Pressure-Low	$\geq$1870 psig	$\geq$1859 psig
10.	Pressurizer Pressure-High	$\leq$2380 psig	$\leq$2391 psig
11.	Pressurizer Water Level-High	$\leq$92% of instrument span	$\leq$93.8% of instrument span
12.	Loss of Flow	$\geq$90% of loop design flow*	$\geq$88.9% of loop design flow*

* Loop design flow = 94,500 gpm
RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
13.	Steam Generator Water Level Low-Low Barton Transmitter Rosemount Transmitter	 $\geq$27.0% of span $\geq$27.0% of span	 $\geq$26.1% of span $\geq$25.7% of span
14.	Steam/Feedwater Flow Mis-Match Coincident With Steam Generator Water Level Low-Low Barton Transmitter Rosemount Transmitter	$\leq$40% of full steam flow at RTP $\geq$27.0% of span $\geq$27.0% of span	$\leq$42.5% of full steam flow at RTP $\geq$26.1% of span $\geq$25.7% of span
15.	Undervoltage - Reactor Coolant Pump	$\geq$4830 volts	$\geq$4760 volts
16.	Underfrequency - Reactor Coolant Pumps	$\geq$57.5 Hz	$\geq$57.1 Hz
17.	Turbine Trip A. Low Trip System Pressure B. Turbine Stop Valve Closure	 $\geq$800 psig $\geq$1% open	 $\geq$750 psig $\geq$1% open

RTP - RATED THERMAL POWER

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TABLE2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
18.	Safety Injection Input from ESF	NA	NA
19.	Reactor Trip System Interlocks		
	A. Intermediate Range Neutron Flux, P-6	$\geq 7.5 \times 10^{-6}$% indication	$\geq 4.5 \times 10^{-6}$% indication
	B. Low Power Reactor Trips Block, P-7 a. P-10 input	$\leq$10% of RTP	$\leq$12.2% of RTP
	b. P-13 input	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% of turbine impulse pressure equivalent
	C. Power Range Neutron Flux P-8	$\leq$38% of RTP	$\leq$40.2% of RTP
	D. Low Setpoint Power Range Neutron Flux, P-10	$\geq$10% of RTP	$\geq$7.8% of RTP
	E. Turbine Impulse Chamber Pressure, P-13	$\leq$10% turbine impulse pressure equivalent	$\leq$12.2% turbine pressure equivalent
	F. Power Range Neutron Flux, P-9	$\leq$50% of RTP	$\leq$52.2% of RTP
20.	Reactor Trip Breakers	NA	NA
21.	Automatic Actuation Logic	NA	NA

RTP - RATED THERMAL POWER

TABLE 2.2-1 (continued)
REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS
NOTATION

NOTE 1: OVERTEMPERATURE ΔT

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

Where:	ΔT	=	Measured ΔT by RTD Instrumentation
	ΔT_o	≤	Indicated ΔT at RATED THERMAL POWER
	K_1	≤	1.23
	K_2	≥	0.0292/°F
	$\frac{1+\tau_1 S}{1+\tau_2 S}$	=	The funtion generated by the lead-lag controller for T_{avg} dynamic compensation
	τ_1, τ_2	=	Time constants utilized in lead-lag controller for T_{avg}, $\tau_1 \ge 28$ secs., $\tau_2 \le 4$ secs.
	T	=	Average temperature, °F
	T'	≤	Indicated T_{avg} at RATED THERMAL POWER, 572.0°F ≤ T' ≤ 587.4°F
	K_3	≥	0.00161/psi
	P	=	Pressurizer pressure, psig
	P'	≥	2235 psig, Nominal RCS operating pressure
	S	=	Laplace transform operator, sec^{-1}.

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 1: (Continued)

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

(i) for $q_t - q_b$ between -35 percent and +6 percent $f_1(\Delta I) = 0$ where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER.

(ii) for each percent that the magnitude of $q_t - q_b$ exceeds -35 percent, the ΔT trip setpoint shall be automatically reduced by 2.46 percent of its value at RATED THERMAL POWER.

(iii) for each percent that the magnitude of $q_t - q_b$ exceeds +6 percent, the ΔT trip setpoint shall be automatically reduced by 3.29 percent of its value at RATED THERMAL POWER.

NOTE 2: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.2 percent ΔT Span.

NOTE 3: OVERPOWER ΔT

$$\Delta T \le \Delta T_o \left[K_4 - K_5 \frac{(\tau_3 S)}{(1 + \tau_3 S)} T - K_6 \left[T - T'' \right] \right]$$

Where:	ΔT	=	as defined in Note 1
	ΔT_o	=	as defined in Note 1
	K_4	≤	1.078
	K_5	≥	0.02/°F for increasing average temperature and 0 for decreasing average temperature
	$\frac{\tau_3 S}{1 + \tau_3 S}$	=	The function generated by the rate-lag controller for T_{avg} dynamic compensation

TABLE 2.2-1 (continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

NOTATION (continued)

NOTE 3: (continued)

τ_3	=	Time constant utilized in rate-lag controller for T_{avg}, $\tau_3 \geq 10$ secs.
K_6	$\geq$	$0.00198/°F$ for $T > T''$ and $K_6 = 0$ for $T \leq T''$
T	=	as defined in Note 1
T''	$\leq$	Indicated T_{avg} at RATED THERMAL POWER, $572.0°F \leq T'' \leq 587.4°F$
S	=	as defined in Note 1

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip point by more than 2.3 percent ΔT Span.

PAGE 2-11
WAS DELETED PER
AMENDMENT No. 28
THAT WAS ISSUED ON
OCTOBER 12, 1984

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

2.1 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 relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio (DNBR) defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability 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.

In meeting this design basis, uncertainties in plant operating parameters, nuclear and thermal parameters, and fuel fabrication parameters are considered statistically such that there is at least a 95 percent probability with 95 percent confidence level that the minimum DNBR for the limiting rod is greater than or equal to the DNBR limit. The uncertainties in the above plant parameters are used to determine the plant DNBR uncertainty. This DNBR uncertainty, combined with the correlation DNBR limit, establishes a design DNBR value which must be met in plant safety analyses using values of input parameters without uncertainties. In addition, margin has been maintained in the design by meeting safety analysis DNBR limits in performing safety analyses.

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

REACTOR CORE (Continued)

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.56 (includes measurement uncertainty) and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} = 1.56\ [1+ 0.3\ (1-P)]$$

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 assuming the axial power imbalance is within the limits of the f_1 (delta I) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature delta T trips will reduce the setpoints to provide protection consistent with core safety limits.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plants, 1971 Edition which permits a maximum transient pressure of 110% (2735 psig) of design pressure. The Reactor Coolant System piping, valves and fittings, are also designed to Section III of the ASME Code for Nuclear Power Plants, 1971 Edition which permits a maximum transient pressure of 120% (2985 psig) of component design pressure. The Safety Limit of 2735 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3107 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS

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

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the reactor trip setpoints have been specified in Table 2.2-1. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

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

REACTOR TRIP SYSTEM INSTRUMENTATION SETPOINTS (Continued)

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

Manual Reactor Trip

The Reactor Protection System includes manual reactor trip capability.

Power Range, Neutron Flux

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

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

Power Range, Neutron Flux, High Rates

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

Intermediate and Source Range, Nuclear Flux

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

Intermediate and Source Range, Nuclear Flux (Continued)

uncontrolled rod cluster control assembly bank withdrawal from a subcritical condition. These trips provide redundant protection to the low setpoint trip of the Power Range, Neutron Flux channels. The Source Range channels will initiate a reactor trip at about 10^{+5} counts per second unless manually blocked when P-6 becomes active. The purpose of the P-6 setpoint, which is above the lower end of the intermediate range scale, is to give the operators sufficient time to actuate the source range reactor trip block. The Intermediate Range channels will initiate a reactor trip at approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active.

Overtemperature ΔT

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 2 seconds) plus thermal delays associated with the RTD's mounted in the thermowells (about 5 seconds), and pressure is within the range between the Pressurizer high and low pressure trips. The setpoint is automatically varied with 1) coolant temperature to correct for temperature induced changes in density and heat capacity of water and includes dynamic compensation for piping delays from the core to the loop temperature detectors, 2) pressurizer pressure, and 3) axial power distribution. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Overpower ΔT

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

Pressurizer Pressure

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

Pressurizer Pressure (Continued)

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

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

Pressurizer Water Level

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

Loss of Flow

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

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

Steam Generator Water Level

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

Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level

The steam/feedwater flow mismatch in coincidence with a steam generator low water level trip is not used in the transient and accident analyses but is included in Table 2.2-1 to ensure the functional capability of the specified trip settings and thereby enhance the overall reliability of the Reactor Protection System. This trip is redundant to the Steam Generator Water Level Low-Low trip. The Steam/Feedwater Flow Mismatch portion of this trip is activated when the steam flow exceeds the feedwater flow by greater than or equal to 40% of full steam flow at RTP. The Steam Generator Low Water level portion of the trip is activated when the water level drops below the low

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

level setpoint, as indicated by the narrow range instrument. These trip values include sufficient allowance in excess of normal operating values to preclude spurious trips but will initiate a reactor trip before the steam generators are dry. Therefore, the required capacity and starting time requirements of the auxiliary feedwater pumps are reduced and the resulting thermal transient on the Reactor Coolant System and steam generators is minimized.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump Bus trips provide reactor core protection against DNB as a result of complete loss of forced coolant flow. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 1.2 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip set point is reached shall not exceed 0.6 seconds. On decreasing power the Undervoltage and Underfrequency Reactor Coolant Pump Bus trips are automatically blocked by P-7 (a power level of approximately 10 percent of RATED THERMAL POWER with a turbine impulse chamber pressure at approximately 10 percent of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine Trip initiates a reactor trip. On decreasing power, the reactor trip from the turbine trip is automatically blocked by P-9 (a power level less than or equal to 50% of RATED THERMAL POWER); and on increasing power, reinstated automatically by P-9.

Safety Injection Input from ESF

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

Reactor Trip System Interlocks

The Reactor Trip System Interlocks perform the following functions:

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

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

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

P-9 On increasing power P-9 automatically enables reactor trip on turbine trip. On decreasing power P-9 automatically blocks reactor trip on turbine trip.

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

P-13 Provides input to P-7.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

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

This specification is not applicable in MODES 5 and 6.

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

APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

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

4.0.3 If it is discovered that a Surveillance was not performed within its specified frequency, as defined by Specification 4.0.2, then compliance with the requirement to declare the LCO not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified frequency, whichever is greater. This delay period is permitted to allow performance of the surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and the risk impact shall be managed.

If the Surveillance is not performed within the delay period, the LCO must immediately be declared not met, and the applicable Action(s) must be entered.

When the Surveillance is performed within the delay period and the Surveillance is not met, the LCO must immediately be declared not met, and the applicable Action(s) must be entered.

4.0.4 Entry into an OPERATIONAL MODE or other specified condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the stated surveillance interval or as otherwise specified. 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 Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

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

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (continued)

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

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - MODES 1 AND 2

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, and 2*.

ACTION:

With the SHUTDOWN MARGIN less than 1.77% delta k/k, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 1.77% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that control bank withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2 with K_{eff} less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

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

*See Special Test Exception 3.10.1

SURVEILLANCE REQUIREMENTS (Continued)

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

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

The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

SHUTDOWN MARGIN - MODES 3, 4 AND 5

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to the limits shown in Figure 3.1-3.

APPLICABILITY: MODES 3, 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

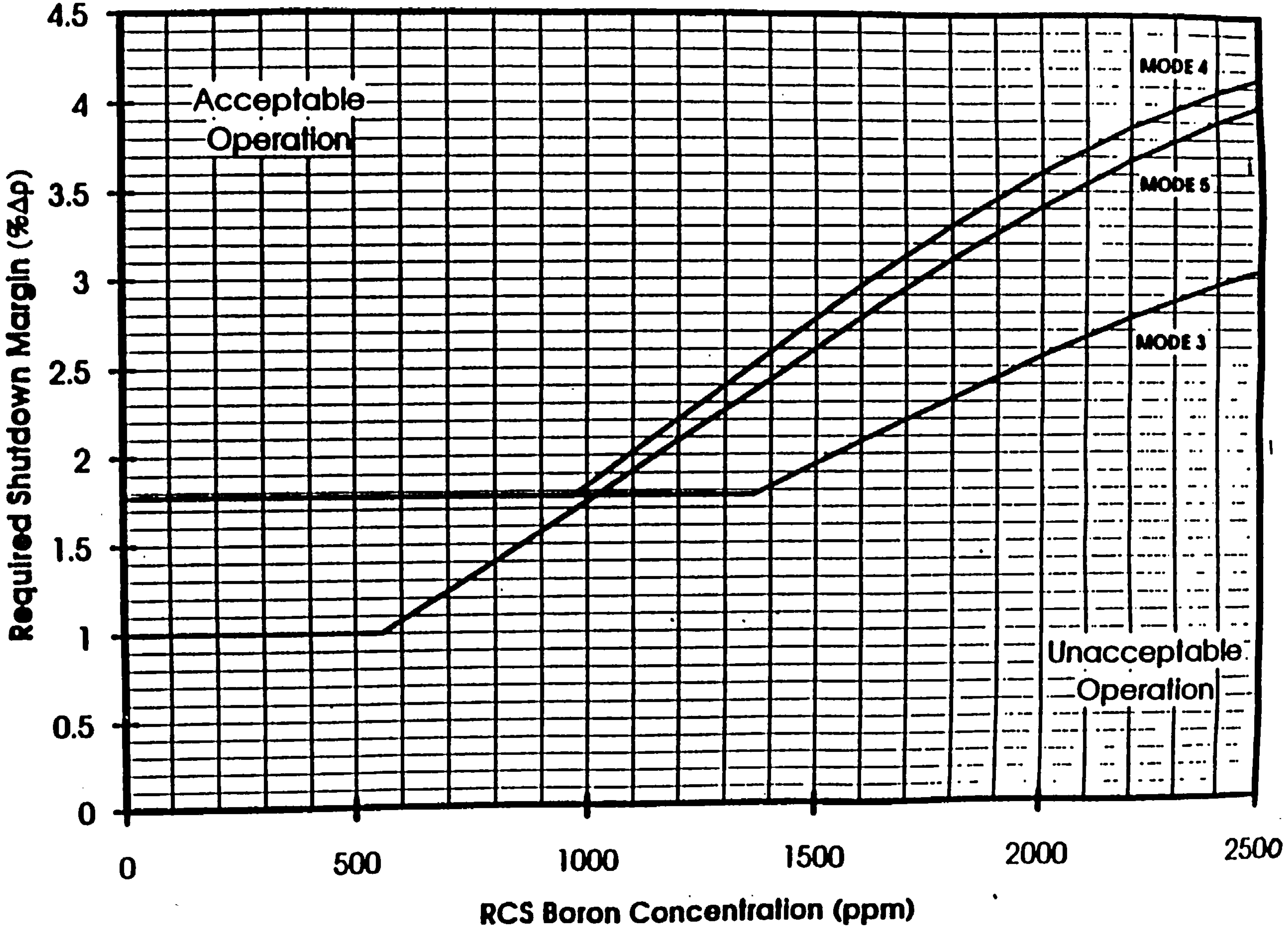
4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to the required value:

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

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

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

FIGURE 3.1-3
REQUIRED SHUTDOWN MARGIN
(MODES 3, 4, AND 5)



MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limit shall be less than or equal to that shown in Figure 3.1-0:

APPLICABILITY: Beginning of Cycle Life (BOL) Limit - MODES 1 and 2* only#
End of Cycle Life (EOL) Limit - MODES 1, 2 and 3 only#

ACTION:

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

   1. Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive than the BOL limit specified in the COLR within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of Specification 3.1.3.6.

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

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

b. With the MTC more negative than the EOL limit specified in the COLR, be in HOT SHUTDOWN within 12 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

b. The MTC shall be measured at any THERMAL POWER and compared to the 300 ppm surveillance limit specified in the COLR (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 the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

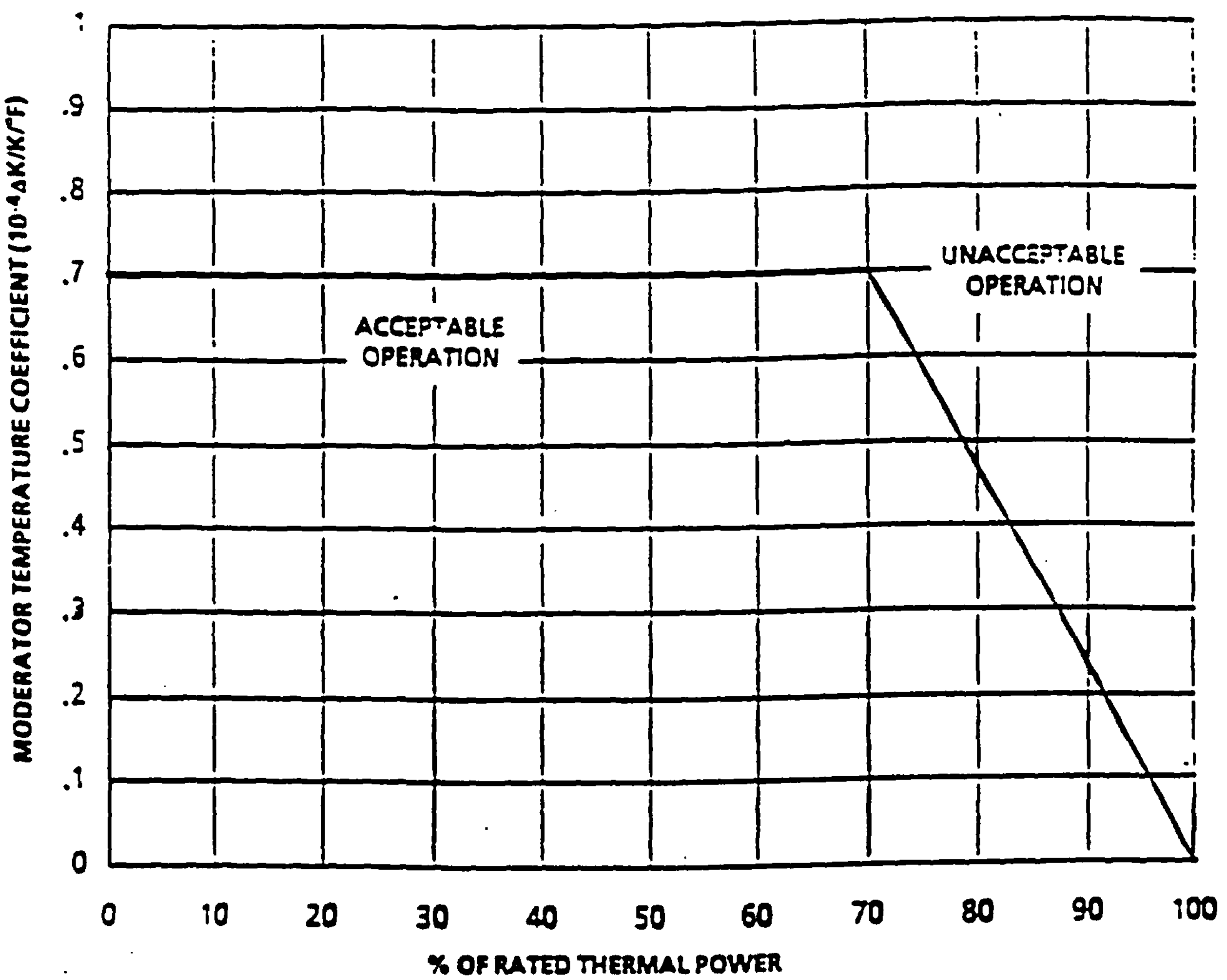


FIGURE 3.1-0
MODERATOR TEMPERATURE COEFFICIENT VS POWER LEVEL

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2#*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.1.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

4.1.2.1.2 Demonstrate operability of the required charging pump per Surveillance 4.5.2.f.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

#Only one boron injection flow path is required to be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

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BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system with:

1. A minimum contained borated water volume of 2700 gallons,
2. Between 7000 and 7700 ppm of boron, and
3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

1. A minimum contained borated water volume of 51,500 gallons,
2. A minimum boron concentration of 2300 ppm, and
3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

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

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

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. A boric acid storage system with:

1. A minimum contained borated water volume of 14,000 gallons,

2. Between 7000 and 7700 ppm of boron, and

3. A minimum solution temperature of 65°F.

b. The refueling water storage tank with:

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

2. A minimum boron concentration of 2300 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the boric acid storage system inoperable and being used as one of the above required borated water sources, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2 percent delta k/k at 200°F; restore the boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.6 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

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

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

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

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERTION

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

APPLICABILITY: MODES 1* and 2*

ACTION:

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

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

c. With more than one full length rod inoperable due to a rod control urgent failure alarm or obvious electrical problem in the rod control system for greater than 72 hours, be in HOT STANDBY within the following 6 hours.

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

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

    2. The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits specified in the CORE OPERATING LIMITS REPORT (COLR); the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation, or

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

*See Special Test Exceptions 3.10.2 and 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

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

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

c) A core power distribution measurement is obtained and $F_Q(z)$ and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

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

SURVEILLANCE REQUIREMENTS

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

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

LIMITING CONDITION FOR OPERATION (Continued)

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.1-1

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

Rod Cluster Control Assembly Insertion Characteristics

Rod Cluster Control Assembly Misalignment

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

Single Rod Cluster Control Assembly Withdrawal At Full Power

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

Major Secondary System Pipe Rupture

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

REACTIVITY CONTROL SYSTEMS

POSITION INDICATION SYSTEMS-OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

   2. Reduce THERMAL POWER TO less than 50% of RATED THERMAL POWER within 8 hours.

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

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

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

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

POSITION INDICATION SYSTEM-SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) rod drop time from the fully withdrawn position shall be less than or equal to 2.7 seconds from beginning of decay of stationary gripper coil voltage to dashpot entry with:

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

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. At least once per 18 months.

SHUTDOWN ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

APPLICABILITY: MODES 1* and 2*#

ACTION:

With a maximum of one shutdown rod inserted beyond the insertion limit specified in the COLR, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Restore the rod to within the limit specified in the COLR, 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 within the insertion limit specified in the COLR.

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

b. At least once per 12 hours thereafter.

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

CONTROL ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The control banks shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled Rod Group Insertion Limits versus Thermal Power For Three Loop Operation.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

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

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

b. Reduce THERMAL POWER within two hours to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the bank position using the insertion limits specified in the COLR, or

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

SURVEILLANCE REQUIREMENTS

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

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

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within:

a. the allowed operational space as specified in the CORE OPERATING LIMITS REPORT (COLR) for Relaxed Axial Offset Control (RAOC) operation, or

b. within the target band specified in the COLR about the target flux difference during base load operation.

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

ACTION:

a. For RAOC operation with the indicated AFD outside of the applicable limits specified in the COLR,

   1. Either restore the indicated AFD to within the COLR specified limits within 15 minutes, or

   2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux - High Trip setpoints to less than or equal 55% of RATED THERMAL POWER within the next 4 hours.

b. For Base Load operation above APL^{ND}** with the indicated AFD outside of the applicable target band about the target flux differences:

   1. Either restore the indicated AFD to within the COLR specified target band within 15 minutes, or

   2. Reduce THERMAL POWER to less than APL^{ND} of RATED THERMAL POWER and discontinue Base Load operation within 30 minutes.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD is within the applicable RAOC limits.

*See Special Test Exception 3.10.2

**APL^{ND} is the minimum allowable power level for base load operation and will be specified in the CORE OPERATING LIMITS REPORT per Specification 6.9.1.11.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AFD shall be determined to be within its limits during POWER OPERATION above 50% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel at least once per 7 days when the AFD Monitor Alarm is OPERABLE:

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 AFD Monitor Alarm 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 limits when two or more OPERABLE excore channels are indicating the AFD to be outside the limits.

4.2.1.3 When in Base Load operation, 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 When in Base Load operation, the target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference in conjunction with the surveillance requirements of Specification 4.2.1.3 above or by linear interpolation between the most recently measured value and the calculated value at the end of cycle life. The provisions of Specification 4.0.4 are not applicable.

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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:

$$F_Q(z) \leq \frac{\left[F_Q^{RTP}\right]}{P}\left[K(z)\right] \text{ for } P > 0.5$$

$$F_Q(z) \leq \left[\frac{F_Q^{RTP}}{0.5}\right]\left[K(z)\right] \text{ for } P \leq 0.5$$

where F_Q^{RTP} = the F_Q limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT (COLR),

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

K(z) = the normalized $F_Q(z)$ for a given core height specified in the COLR.

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 delta T Trip Setpoints have been reduced at least 1% for each 1% $F_Q(z)$ exceeds the limit.

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

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 For RAOC operation, $F_Q(z)$ 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

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the Measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER, and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{P \times W(z)} \text{ for } P > 0.5$$

$$F_Q^M(z) \leq \frac{F_Q^{RTP} \times K(z)}{W(z) \times 0.5} \text{ for } P \leq 0.5$$

where $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR, F_Q^{RTP} is the F_Q limit, K(z) is the normalized $F_Q(z)$ as a function of core height, P is the relative THERMAL POWER, and W(z) is the cycle dependent function that accounts for power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and W(z) are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

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

1. Upon achieving equilibrium conditions after exceeding by 10% or more of RATED THERMAL POWER, the THERMAL POWER at which $F_Q(z)$ was last determined, * or

2. At least once per 31 Effective Full Power Days, whichever occurs first.

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* 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 the core power distribution measurement is obtained.

SURVEILLANCE REQUIREMENTS (Continued)

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. Increase $F_Q^M(z)$ by the appropriate penalty factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.2.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until two successive core power distribution measurements indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationships specified in Specification 4.2.2.2.c. above not being satisfied:

1. Calculate the maximum percent over the core height (z) that $F_Q(z)$ exceeds its limit by the following expression:

$$\left\{\left[\frac{F_Q^M(z) x W(z)}{\frac{F_Q^{RTP}}{P} x K(z)}\right]-1\right\} x\ 100 \text{ for } P \geq 0.5$$

$$\left\{\left[\frac{F_Q^M(z) x W(z)}{\frac{F_Q^{RTP}}{0.5} x K(z)}\right]-1\right\} x\ 100 \text{ for } P < 0.5$$

SURVEILLANCE REQUIREMENTS (Continued)

2. One of the following actions shall be taken:

   (a) Within 15 minutes, control the AFD to within new AFD limits which are determined by reducing the applicable AFD limits by 1% AFD for each percent $F_Q(z)$ exceeds its limits as determined in Specification 4.2.2.2.f.(1). Within 8 hours, reset the AFD alarm setpoints to these modified limits, or

   (b) Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the percent calculated above, or

   (c) Verify that the requirements of Specification 4.2.2.3 for Base Load operation are satisfied and enter Base Load operation.

g. The limits specified in Specifications 4.2.2.2.c., 4.2.2.2.e., and 4.2.2.2.f. above are not applicable in the following core plane regions:

1. Lower core region from 0 to 10%, inclusive.
2. Upper core region from 90 to 100%, inclusive.

4.2.2.3 Base Load operation is permitted at powers above APL^{ND} if the following conditions are satisfied:

a. Prior to entering Base Load operation, maintain THERMAL POWER above APL^{ND} and less than or equal to that allowed by Specification 4.2.2.2 for at least the previous 24 hours. Maintain Base Load operation surveillance (AFD within applicable target band about the target flux difference) during this time period. Base Load operation is then permitted providing THERMAL POWER is maintained between APL^{ND} and APL^{BL} or between APL^{ND} and 100% (whichever is most limiting) and F_Q surveillance is maintained pursuant to Specification 4.2.2.4. APL^{BL} is defined as the minimum value of:

$$APL^{BL} = \frac{F_Q^{RTP} \times K(z)}{F_Q^M(z) \times W(z)_{BL}} \times 100\%$$

over the core height (z) where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing tolerances and measurement uncertainty as specified in the COLR. The F_Q limit is F_Q^{RTP}. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transient encountered during base load operation. F_Q^{RTP}, K(z), and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

b. During Base Load operation, if the THERMAL POWER is decreased below APL^{ND} then the conditions of 4.2.2.3.a shall be satisfied before re-entering Base Load operation.

4.2.2.4 During Base Load Operation $F_Q(z)$ 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 above APL^{ND} when the Power Distribution Monitoring System (PDMS) is inoperable; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. Using the PDMS at any THERMAL POWER greater than APL^{ND}; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

c. Satisfying the following relationship:

$$F_Q^M(z) \le \frac{F_Q^{RTP} \times K(z)}{P \times W(z)_{BL}} \text{ for } P > APL^{ND}$$

where: $F_Q^M(z)$ is the measured $F_Q(z)$ increased by the applicable allowances for manufacturing and measurement uncertainties as specified in the COLR. The F_Q limit is F_Q^{RTP}. P is the relative THERMAL POWER. $W(z)_{BL}$ is the cycle dependent function that accounts for limited power distribution transients encountered during normal operation. F_Q^{RTP}, K(z) and $W(z)_{BL}$ are specified in the CORE OPERATING LIMITS REPORT as per Specification 6.9.1.11.

d. Measuring $F_Q^M(z)$ in conjunction with target flux difference determination according to the following schedule:

1. Prior to entering BASE LOAD operation after satisfying Section 4.2.2.3 unless a core power distribution measurement has been obtained in the previous 31 EFPD with the relative thermal power having been maintained above APL^{ND} for the 24 hours prior to measurement, and

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

e. With the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) increasing since the previous determination of $F_Q^M(z)$ either of the following actions shall be taken:

1. Increase $F_Q^M(z)$ by the appropriate penalty factor specified in the COLR and verify that this value satisfies the relationship in Specification 4.2.2.4.c, or

2. $F_Q^M(z)$ shall be measured at least once per 7 Effective Full Power Days until 2 successive core power distribution measurements indicate that the maximum value of

$$\frac{F_Q^M(z)}{K(z)}$$

over the core height (z) is not increasing.

f. With the relationship specified in 4.2.2.4.c above not being satisfied, either of the following actions shall be taken:

1. Place core in an equilibrium condition where the limit in 4.2.2.2.c is satisfied, and remeasure $F_Q^M(z)$, or

2. Comply with the requirements of Specification 3.2.2 for $F_Q(z)$ exceeding its limit by the maximum percent calculated over the core height (z) with the following expression:

$$\left\{\left[\frac{F_Q^M(z) \times W(z)_{BL}}{\frac{F_Q^{RTP}}{P} \times K(z)}\right]-1\right\} \times 100 \text{ for } P \geq APL^{ND}$$

g. The limits specified in 4.2.2.4.c, 4.2.2.4.e, and 4.2.2.4.f above are not applicable in the following core plane regions:

1. Lower core region 0 to 10%, inclusive.

2. Upper core region 90 to 100%, inclusive.

4.2.2.5 When $F_Q(z)$ is measured for reasons other than meeting the requirements of Specification 4.2.2.2 an overall measured $F_Q(z)$ shall be obtained:

a. From a power distribution map

1. When THERMAL POWER is ≤ 25%, but > 5% of RATED THERMAL POWER, or

2. When the Power Distribution Monitoring System (PDMS) is inoperable;

and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

b. From the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER; and increasing the measured $F_Q(z)$ by the applicable manufacturing and measurement uncertainties as specified in the COLR.

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

LIMITING CONDITION FOR OPERATION

3.2.3 The combination of indicated Reactor Coolant System (RCS) total flow rate and R shall be maintained within the region of allowable operation as specified in the CORE OPERATING LIMITS REPORT (COLR) figure entitled RCS Total Flow Rate Versus R For Three Loop Operation.

Where:

a. R = $\dfrac{F^N_{\Delta H}}{F^{RTP}_{\Delta H}\left[1.0+PF_{\Delta H}(1.0-P)\right]}$,

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

c. $F^N_{\Delta H}$ = Measured values of $F^N_{\Delta H}$ obtained by

1. Using the movable incore detectors to obtain a power distribution map when THERMAL POWER is ≤ 25% but > 5% of RATED THERMAL POWER, or when PDMS is inoperable, and

2. Using the PDMS when THERMAL POWER is > 25% of RATED THERMAL POWER.

The measured values of $F^N_{\Delta H}$ shall be increased by the applicable $F^N_{\Delta H}$ measurement uncertainties as specified in the COLR, and used to calculate R since the RCS Total Flow Rate Versus R figure in the COLR includes measurement uncertainties of 2.1% (includes 0.1% for feedwater venturi fouling) for flow,

d. $F^{RTP}_{\Delta H}$ = The $F^N_{\Delta H}$ limit at RATED THERMAL POWER specified in the COLR, and

e. $PF_{\Delta H}$ = The Power Factor Multiplier specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the combination of RCS total flow rate and R outside the region of acceptable operation specified in the COLR:

a. Within 2 hours either:

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

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

b. Within 24 hours of initially being outside the above limits, verify through a core power distribution measurement and RCS total flow rate comparison that the combination of R and RCS total flow rate are restored to within the above limits, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

ACTION: (Continued)

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION items a.2. and/or b. above; subsequent POWER OPERATION may proceed provided that the combination of R and indicated RCS total flow rate are demonstrated, through a core power distribution measurement and RCS total flow rate comparison, to be within the region of acceptable operation specified in the COLR prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,

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

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

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The combination of indicated RCS total flow rate and R shall be determined to be within the region of acceptable operation specified in the COLR.

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

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

4.2.3.3 The indicated RCS total flow rate shall be verified to be within the region of acceptable operation specified in the COLR at least once per 12 hours when the most recently obtained value of R, obtained per Specification 4.2.3.2, is assumed to exist.

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

4.2.3.5 The RCS total flow rate shall be determined by heat balance measurement at ≥90% RATED THERMAL POWER at least once per 18 months.

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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 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

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

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

*See Special Test Exception 3.10.2.

ACTION: (Continued)

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

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

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

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

2. Reduce THERMAL POWER at least 3% from RATED THERMAL POWER for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0, within 30 minutes.

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

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

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

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

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

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

ACTION: (Continued)

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a full-core flux map, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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

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

4.2.4.2 The QUADRANT POWER TILT RATIO shall be determined to be within the limit when above 75 percent of RATED THERMAL POWER with one Power Range Channel inoperable by using the movable incore detectors to confirm that the normalized symmetric power distribution, obtained from 2 sets of 4 symmetric thimble locations or a core power distribution measurement, is consistent with the indicated QUADRANT POWER TILT RATIO at least once per 12 hours.

3/4 2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.2-1

DNB PARAMETERS

PARAMETER	LIMITS: 3 Loops In Operation	LIMITS: 2 Loops In Operation
Indicated Reactor Coolant System T_{avg}	$\leq$ 589.2°F	**
Indicated Pressurizer Pressure	$\geq$ 2206 psig*	**

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

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

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1	6#
12. A. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop each operating loop	1	6#
13. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	6#
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	2/loop-level and 2/loop-flow mismatch in each loop	1/loop-level coincident with 1/loop-flow mismatch in same loop	1/loop-level and 2/loop-flow mismatch in same loop or 2/loop-level and 1/loop-flow mismatch in same loop	1, 2	6#

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

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

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
15. Undervoltage-Reactor Coolant Pumps	3-1/bus	2	2	1	$6^{\#}$
16. Underfrequency-Reactor Coolant Pumps	3-1/bus	2	2	1	$6^{\#}$
17. Turbine Trip					
A. Low Fluid Oil Pressure	3	2	2	1	$6^{\#}$
B. Turbine Stop Valve Closure	4	4	1	1	$10^{\#}$
18. Safety Injection Input from ESF	2	1	2	1, 2	12

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
19. Reactor Trip System Interlocks					
A. Intermediate Range Neutron Flux, P-6	2	1	2	2##	7
B. Low Power Reactor Trips Block, P-7 — P-10 Input	4	2	3	1	7
P-13 Input	2	1	2	1	7
C. Power Range Neutron Flux, P-8	4	2	3	1	7
D. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
E. Turbine First Stage Pressure, P-13	2	1	2	1	7
F. Power Range Neutron Flux, P-9	4	2	3	1	7
20. Reactor Trip Breakers	2	1	2	1, 2	8, 11
	2	1	2	3*, 4*, 5*	9
21. Automatic Trip Logic	2	1	2	1, 2	12
	2	1	2	3*, 4*, 5*	9

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

#The provisions of Specification 3.0.4 are not applicable.

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

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

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

ACTION STATEMENTS

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

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

a. The inoperable channel is placed in the tripped condition within 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

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

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

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

ACTION 6 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours; and

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.1.1.

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

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

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

ACTION 10 - With the number of OPERABLE Channels less than the Total Number of Channels operation may continue provided the inoperable channels are placed in the tripped condition within 6 hours.

ACTION 11 - With one of the diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status within 48 hours or declare the breaker inoperable and apply ACTION 8. The breaker shall not be bypassed while one of the diverse trip features is inoperable except for the time required for performing maintenance to restore the breaker OPERABLE status.

ACTION 12 With the number of OPERABLE Channels one less than the minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 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.

TABLE 3.3-2

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

	FUNCTIONAL UNIT	RESPONSE TIME
1.	Manual Reactor Trip	Not Applicable
2.	Power Range, Neutron Flux	$\leq$ 0.5 seconds(1)
3.	Power Range, Neutron Flux, High Positive Rate	Not Applicable
4.	Deleted	
5.	Intermediate Range, Neutron Flux	Not Applicable
6.	Source Range, Neutron Flux	$\leq$ 0.5 seconds(1)
7.	Overtemperature ΔT	$\leq$ 8.5 seconds(1)(2)
8.	Overpower ΔT	$\leq$ 8.5 seconds(1)(2)
9.	Pressurizer Pressure--Low	$\leq$ 2.0 seconds
10.	Pressurizer Pressure--High	$\leq$ 2.0 seconds
11.	Pressurizer Water Level--High	Not Applicable

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

(2) The 8.5 second response time includes a 5.0 second delay for the RTDs mounted in thermowells.

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. A. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
B. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
13. Steam Generator Water Level--Low-Low	≤ 2.0 seconds
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	Not Applicable
15. Undervoltage-Reactor Coolant Pumps	≤ 1.5 seconds
16. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
17. Turbine Trip	
A. Low Fluid Oil Pressure	Not Applicable
B. Turbine Stop Valve Closure	Not Applicable
18. Safety Injection Input from ESF	Not Applicable
19. Reactor Trip System Interlocks	Not Applicable
20. Reactor Trip Breakers	Not Applicable
21. Automatic Trip Logic	Not Applicable

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

Correction letter of 8-6-91

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
D. Low Setpoint Power Range Neutron Flux, P-10	N.A.	R(4)	R	N.A.	N.A.	1, 2
E. Turbine Impulse Chamber Pressure, P-13	N.A.	R	R	N.A.	N.A.	1
F. Low Power Range Neutron Flux, P-9	N.A.	R(4)	R	N.A.	N.A.	1
20. Reactor Trip Breaker	N.A.	N.A.	N.A.	M (7, 12)	N.A.	1, 2, 3*, 4*, 5*
21. Automatic Trip Logic	N.A.	N.A.	N.A.	N.A.	M (7)	1, 2, 3*, 4*, 5*
22. Reactor Trip Bypass Breaker	N.A.	N.A.	N.A.	M(13), R(14)	N.A.	1, 2, 3*, 4*, 5*

Correction Letter of 8-6-91

TABLE 4.3-1 (Continued)

TABLE NOTATION

* -	With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.
## -	Below P-6 (Intermediate Range Neutron Flux Interlock) setpoint.
### -	Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.
(1) -	If not performed in previous 31 days.
(2) -	Comparison of calorimetric to excore power indication above 15% of RATED THERMAL POWER. Adjust excore channel gains consistent with calorimetric power if absolute difference is greater than 2 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(3)	Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3 percent. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(4) -	Neutron detectors may be excluded from CHANNEL CALIBRATION.
(5) -	Detector plateau curves shall be obtained evaluated and compared to manufacturer's data. For the Power Range Neutron Flux Channels the provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(6) -	Incore - Excore Calibration, above 75% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.
(7) -	Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(8) -	DELETED
(9) -	Quarterly Surveillance in MODES 3*, 4* and 5* shall also include verification that permissives P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window.
(10) -	Setpoint verification is not required.
(11) -	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).
(12) -	The TRIP ACTUATING DEVICE OPERATIONAL TEST shall independently verify the OPERABILITY of the undervoltage and shunt trip attachments of the Reactor Trip Breakers.
(13) -	Local manual shunt trip prior to placing breaker in service.
(14) -	Automatic undervoltage trip.

3/4 3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

a. With an ESFAS Instrumentation or Interlock 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 3.3-4, adjust the Setpoint consistent with the Trip Setpoint value.

b. With an ESFAS Instrumentation or Interlock Trip Setpoint less conservative than the value shown in the Allowable Value Column of Table 3.3-4, declare the channel inoperable and apply the applicable ACTION statement requirements of Table 3.3-3 until the channel is restored to its OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

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

SURVEILLANCE REQUIREMENTS

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

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

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TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION, REACTOR TRIP, FEEDWATER ISOLATION, CONTROL ROOM ISOLATION, START DIESEL GENERATORS, CONTAINMENT COOLING FANS AND ESSENTIAL SERVICE WATER.					
a. Manual Initiation	2	1	2	1, 2, 3, 4	18
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c. Reactor Building Pressure - High-1	3	2	2	1, 2, 3	24*
d. Pressurizer Pressure - Low	3	2	2	1, 2, 3#	24*
e. Differential Pressure Between Steam Lines - High	3/steam line	2/steam line twice and 1/3 steam lines	2/steam line	1, 2, 3	24*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT		TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
f.	Steam Line Pressure-Low	1 pressure/ loop	1 pressure and 2 loops	1 pressure and 2 loops	1, 2, 3##	24*
2.	REACTOR BUILDING SPRAY					
a.	Manual	2 sets - 2 switches/set	1 set	2 sets	1, 2, 3, 4	18
b.	Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
c.	Reactor Building Pressure--High-3 (Phase 'A' isolation aligns spray system discharge valves and NaOH tank suction valves)	4	2	3	1, 2, 3	16

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. CONTAINMENT ISOLATION					
a. Phase "A" Isolation					
1) Manual	2	1	2	1, 2, 3, 4	18
2) Safety Injection	See 1 above for all safety injection initiating functions and requirements.				
3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
b. Phase "B" Isolation					
1) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	14
2) Reactor Building Pressure--High-3	4	2	3	1, 2, 3	16
c. Purge and Exhaust Isolation					
1) Safety Injection	See 1 above for all safety injection initiating functions and requirements.				
2) Containment Radio-activity- High	2*	1	2*	1, 2, 3, 4	17
3) Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3, 4	17

*Purge exhaust monitor not required when purge exhaust is closed.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. STEAM LINE ISOLATION					
a. Manual					
i. One Switch/line	1/steam line	1/steam line	1/operating steam line	1, 2, 3###	23
ii. One Switch/all lines	1	1	1	1, 2, 3###	23
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3###	21
c. Reactor Building Pressure--High-2	3	2	2	1, 2, 3###	24*
d. Steam Flow in Two Steam Lines--High	2/steam line	1/steam line any 2 steam lines	1/steam line	1, 2, 3###	24*
COINCIDENT WITH T_{avg}--Low-Low	1 T_{avg}/loop	1 T_{avg} any 2 loops	1 T_{avg} any 2 loops	1, 2, 3###	24*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
e. Steam Line Pressure-Low	1 pressure/ loop	1 pressure any 2 loops	1 pressure any 2 loops	1, 2, 3##,###	24*
5. TURBINE TRIP & FEEDWATER ISOLATION					
a. Steam Generator Water Level--High-High	3/loop	2/loop in any operating loop	2/loop in each operating loop	1, 2	24*

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. EMERGENCY FEEDWATER					
a. Manual Initiation	1 per pump	1 per pump	1 per pump	1, 2, 3	22
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
c. Stm. Gen. Water Level-Low-Low					
i. Start Motor-Driven Pumps	3/stm. gen.	2/stm. gen. any stm gen.	2/stm. gen.	1, 2, 3	24*
ii. Start Turbine-Driven Pump	3/stm. gen.	2/stm. gen. any 2 stm. gen.	2/stm. gen	1, 2, 3	24*
d. Undervoltage-both ESF Busses Start Turbine-Driven Pump	2-1/bus	2	2	1, 2, 3	19
e. S.I. Start Motor-Driven Pumps	See 1 above (all S.I. initiating functions and requirements)				
f. Undervoltage-one ESF bus Start Motor-Driven Pumps	2-1/bus	1	2	1, 2	22
g. Trip of Main Feedwater Pumps Start Motor-Driven Pumps	3-1/pump	3-1/pump	3-1/pump	1, 2	19
h. Suction Transfer on Low Pressure	4	2	3	1, 2, 3	16

Correction letter of 8-6-91

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. LOSS OF POWER					
a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	2-1/bus	1	2	1, 2, 3, 4	18
b. 7.2 kv Emergency Bus Undervoltage (Degraded Voltage)	2-1/bus	1	2	1, 2, 3, 4	18
8. AUTOMATIC SWITCHOVER TO CONTAINMENT SUMP					
a. RWST level low-low	4	2	3	1, 2, 3	16
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	21
9. ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS					
a. Pressurizer Pressure, P-11	3	2	2	1, 2, 3	20
b. Low-Low T_{avg}, P-12	3	2	2	1, 2, 3###	20
c. Reactor Trip, P-4	2	2	2	1, 2, 3	22

TABLE 3.3-3 (Continued)

TABLE NOTATION

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

\## Trip function may be blocked in this MODE below the P-12 (Low-Low T_{avg} Interlock) setpoint.

\### Except when below P-12 with all MSIVs and bypasses closed and disabled.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 14 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

ACTION 15 - DELETED

ACTION 16 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may continue provided the inoperable channel is placed in bypass and the Minimum Channels OPERABLE requirement is met. Restore the inoperable channel to OPERABLE status within 6 hours otherwise;

Be in at least HOT STANDBY within the next 6 hours <u>and</u> in COLD SHUTDOWN within the following 30 hours.

One additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1.

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

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

ACTION 19 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

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

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

TABLE 3.3-3 (Continued)

ACTION STATEMENTS (Continued)

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

ACTION 21 - With the number of OPERABLE Channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channels to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1 provided the other channel is OPERABLE.

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

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

ACTION 24 - 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 6 hours.

b. The Minimum Channels OPERABLE requirement is met; however, the inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels per Specification 4.3.2.1.

ACTION 25 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable Channels to OPERABLE status within 6 hours or be in at least HOT STANDBY within the next 6 hours; however, one channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.2.1, provided the other channel is OPERABLE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	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.		
	a. Manual Initiation	NA	NA
	b. Automatic Actuation Logic	NA	NA
	c. Reactor Building Pressure-High 1	≤3.6 psig	≤3.86 psig
	d. Pressurizer Pressure--Low	≥1850 psig	≥1839 psig
	e. Differential Pressure Between Steamlines--High	≤97 psig	≤106 psi
	f. Steamline Pressure--Low	≥675 psig	≥635 psig(1)
2.	REACTOR BUILDING SPRAY		
	a. Manual Initiation	NA	NA
	b. Automatic Actuation Logic and Actuation Relays	NA	NA
	c. Reactor Building Pressure-High 3 (Phase 'A' isolation aligns spray system discharge valves and NaOH tank suction valves)	≤12.05 psig	≤12.31 psig

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

* Trip setpoints shall be set to ensure that the limits of ODCM Specification 1.2.2.1 are not exceeded.

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
4.	STEAM LINE ISOLATION		
	a. Manual	NA	NA
	b. Automatic Actuation Logic and Actuation Relays	NA	NA
	c. Reactor Building Pressure-High 2	$\leq$6.35	$\leq$6.61
	d. Steam Flow in Two Steamlines-High, Concident with	$\leq$ a function defined as follows: A Δp corresponding to 40% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 110% of full steam flow at full load	$\leq$ a function defined as follows: A Δp corresponding to 44% of full steam flow between 0% and 20% load and then a Δp increasing linearly to a Δp corresponding to 114.0% of full steam flow at full load
	T_{avg} - Low-Low	$\geq$552.0°F	$\geq$548.4°F
	e. Steamline Pressure-Low	$\geq$675 psig	$\geq$635 psig(1)

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
5.	TURBINE TRIP AND FEEDWATER ISOLATION		
	a. Steam Generator Water Level - High-High Barton Transmitter Rosemount Transmitter	 ≤79.2% of span ≤79.2% of span	 ≤81.0% of span ≤81.0% of span
	b. Automatic Actuation Logic	NA	NA
6.	EMERGENCY FEEDWATER		
	a. Manual	NA	NA
	b. Automatic Actuation Logic	NA	NA
	c. Steam Generator Water Level - Low-Low Barton Transmitter Rosemount Transmitter	 ≥27.0% of span ≥27.0% of span	 ≥26.1% of span ≥25.7% of span
	d. & f. Undervoltage-ESF Bus	≥ 5760 Volts with a ≤0.25 second time delay ≥ 6576 Volts with a ≤3.0 second time delay	≥ 5652 Volts with a ≤0.275 second time delay ≥ 6511 Volts with a ≤3.3 second time delay

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

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

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

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

	Functional Unit	Trip Setpoint	Allowable Value
9.	ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INTERLOCKS		
	INTERLOCKS		
	a. Pressurizer Pressure, P-11	1985 psig	$\geq$1974 psig & $\leq$1996 psig
	b. T_{avg} Low-Low, P-12	552°F	$\geq$548.4°F & $\leq$555.6°F
	c. Reactor Trip, P-4	NA	NA

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
1.	Manual		
	a.	Safety Injection	Not Applicable
	b.	Reactor Building Spray	Not Applicable
	c.	Containment Isolation Phase "A" Isolation	Not Applicable
	d.	Steam Line Isolation	Not Applicable
	e.	Feedwater Isolation	Not Applicable
	f.	Emergency Feedwater	Not Applicable
	g.	Essential Service Water	Not Applicable
	h.	Reactor Building Cooling Fans	Not Applicable
	i.	Control Room Isolation	Not Applicable
2.	Reactor Building Pressure-High		
	a.	Safety Injection (ECCS)	≤27.0(2)/27.0(1)
	b.	Reactor Trip (from SI)	≤3.0
	c.	Feedwater Isolation	≤10.0
	d.	Containment Isolation-Phase "A"	≤45.0(4)/55.0(5)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION			RESPONSE TIME IN SECONDS
	e.	Reactor Building Purge and Exhaust Isolation	Not Applicable
	f.	Emergency Feedwater Pumps	Not Applicable
	g.	Service Water System	≤71.5(4)/81.5(5)
	h.	Reactor Building Cooling Units	≤76.5(4)/86.5(5)
	i.	Control Room Isolation	Not Applicable
3.	Pressurizer Pressure-Low		
	a.	Safety Injection (ECCS)	≤27.0(2)/27.0(1)
	b.	Reactor Trip (from SI)	≤3.0
	c.	Feedwater Isolation	≤10.0
	d.	Containment Isolation -Phase "A"	≤45.0(4)/55.0(5)
	e.	Reactor Building Purge and Exhaust Isolation	Not Applicable
	f.	Emergency Feedwater Pumps	Not Applicable
	g.	Service Water System	≤71.5(4)/81.5(5)
	h.	Reactor Building Cooling Units	≤76.5(4)/86.5(5)
	i.	Control Room Isolation	Not Applicable
4.	Differential Pressure Between Steam Lines-High		
	a.	Safety Injection (ECCS)	≤27.0(2)/37.0(3)
	b.	Reactor Trip (from SI)	≤3.0
	c.	Feedwater Isolation	≤10.0
	d.	Containment Isolation -Phase "A"	≤45.0(4)/55.0(5)

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
e.	Reactor Building Purge and Exhaust Isolation	Not Applicable
f.	Emergency Feedwater Pumps	Not Applicable
g.	Service Water System	≤71.5(4)/81.5(5)
h.	Reactor Building Cooling Units	≤76.5(4)/86.5(5)
i.	Control Room Isolation	Not Applicable
5.	Steam Line Pressure-Low	
a.	Safety Injection - ECCS	≤27.0(2)/37.0(3)
b.	Reactor Trip (from SI)	≤3.0
c.	Feedwater Isolation	≤10.0
d.	Containment Isolation - Phase "A"	≤45.0(4)/55.0(5)
e.	Reactor Building and Purge and Exhaust Isolation	Not Applicable
f.	Emergency Feedwater Pumps	Not Applicable
g.	Service Water System	≤71.5(4)/81.5(5)
h.	Reactor Building Cooling Units	≤76.5(4)/86.5(5)
i.	Steam Line Isolation	≤10.0
j.	Control Room Isolation	Not Applicable
6.	Steam Flow in Two Steam Lines - High Coincident with Tavg--Low-Low	
a.	Steam Line Isolation	≤12.0
7.	Reactor Building Pressure-High-2	
a.	Steam Line Isolation	≤9.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

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

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION		RESPONSE TIME IN SECONDS
13.	Trip of Main Feedwater Pumps	
	a. Motor-driven Emergency Feedwater Pumps	Not Applicable
14.	Loss of Power	
	a. 7.2 kv Emergency Bus Undervoltage (Loss of Voltage)	$\leq$10.3
	b. 7.2 kv Emergency Bus Undervoltage (Degraded Voltage)	$\leq$13.3
15.	Containment Radioactivity--High	
	a. Purge and Exhaust Isolation	Not Applicable
16.	RWST Level--Low-Low	
	a. Automatic Switchover to Containment Sump	Not Applicable
17.	Aux Feed Suction Pressure Low	
	a. Suction transfer	Not Applicable
Note:	Response time for Motor-driven Emergency Feedwater Pumps on all SI signal starts	$\leq$60.0

TABLE 3.3-5 (Continued)

TABLE NOTATION

(1) Diesel generator starting and sequence loading delays from under voltage included. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps and RHR pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is not included.

(2) Diesel generator starting delay not included. Sequence loading delay included. Offsite power available. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then VCT valves close) is included.

(3) Diesel generator starting and sequence loading delays from under voltage included. Response time limit includes positioning of valves to establish SI path and attainment of discharge pressure for centrifugal charging pumps. Sequential transfer of centrifugal charging pump suction from the VCT to the RWST (RWST valves open, then the VCT valves close) is included.

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

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

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABL. (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

Correction letter of 8-6-91

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

TABLE NOTATION

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

(2) The 36 inch containment purge supply and exhaust isolation valves are sealed closed during Modes 1 through 4, as required by TS 3.6.1.7. With these valves sealed closed, their ability to open is defeated; therefore, they are excluded from the quarterly slave relay test.

INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	1	6	≤ 1 R/hr	1 - 10^{5} mr/hr	28
c. Reactor Building Area					
i. High Range RM-G7 and High Range RM-G18	2	1, 2, 3 & 4	N/A	10 - 10^{7} R/hr 1 - 10^{7} R/hr	30
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	< 1 x 10^{-5} µCi/cc (Kr-85)	10 - 10^{6} cpm	27
ii. Particulate Activity	1	**	N/A	10 - 10^{6} cpm	27
b. Containment					
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	1	6	≤ 2 x background***	10 - 10^{6} cpm	28
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool
*** Alarm/trip setpoint will be per the Operational Dose Calculation Manual when purge exhaust operations are in progress

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Spent Fuel Pool Area (RM-G8)	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	25
b. Reactor Building Manipulator Crane Area (RM-G17A or RM-G17B)	1	6	≤ 1 R/hr	1 - 10^{5} mR/hr	28
2. PROCESS MONITORS					
a. Spent Fuel Pool Exhaust - Ventilation System (RM-A6)					
i. Gaseous Activity	1	**	≤ 1 x 10^{-5} μCi/cc (Kr-85)	10 - 10^{6} cpm	27
ii. Particulate Activity	1	**	N/A	10 - 10^{6} cpm	27
b. Containment					
i. Gaseous Activity - Purge & Exhaust Isolation (RM-A4)	1	6	≤ 2 x background***	10 - 10^{6} cpm	28
ii. Particulate and Gaseous Activity (RM-A2) - RCS Leakage Detection	1	1, 2, 3 & 4	N/A	10 - 10^{6} cpm	26
c. Control Room Isolation (RM-A1)	1	ALL MODES	≤ 2 x background	10 - 10^{6} cpm	29

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

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

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

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

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

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

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

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

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

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

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

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

INSTRUMENTATION

MOVABLE INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The movable incore detection system shall be OPERABLE with:

a. At least 75% of the detector thimbles,

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

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

APPLICABILITY: When the movable incore detection system is used for:

a. Recalibration of the excore neutron flux detection system,

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

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$ and $F_Q(z)$.

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INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

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

Elevations nominal above grade elevation

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

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

Elevations nominal above grade elevation

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

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

CREP - Control Room Evacuation Panel

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N. A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Pressure	M	R
5. Steam Generator Level	M	R
6. Condensate Storage Tank Level	M	R
7. Reactor Coolant System Hot Leg Temperature	M	R
8. Reactor Coolant System Cold Leg Temperature	M	R
9. Reactor Coolant System Pressure	M	R
10. Pressurizer Relief Tank Level	M	R
11. Reactor Building Temperature	M	R
12. Boric Acid Tank Level	M	R

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-10; either restore the inoperable channels to OPERABLE status witin 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown on Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 30 days or submit a Special Report within the following 14 days from the time the action is required. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels to operable status.

b.1 With the number of OPERABLE Reactor Building radiation monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channel(s) to OPERABLE status within 72 hours, or:

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

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

b.2 With the number of Hydrogen monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, restore at least one monitor to operable status within 72 hours or be in at least HOT STANDBY within the next 6 hours, and in HOT SHUTDOWNwithin the next 12 hours.

b.3 With the number of OPERABLE accident monitoring channels less than the Minimum Channels Operable requirement of Table 3.3-10, either restore the inoperable channels to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each accident monitoring instrumentation channel shall be demonstrated OPERABLE by performing a monthly CHANNEL CHECK and a CHANNEL CALIBRATION every refueling outage. The Reactor Building Radiation Level Instrumentation CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for the range decades above 10R/hr and a single point calibration of the detector below 10R/hr with an installed or portable gamma source.

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Building Pressure - Narrow Range Instrument Loop/Indicator: Channel D IPT-951/IPI-951 Channel B IPT-952/IPI-952	2	1
2. Reactor Building Pressure - Wide Range Instrument Loop/Indicator: Channel D IPT-954A/IPI-954A Channel E IPT-954B/IPI-954B	2	1
3. Reactor Building Radiation Level - High Range Instrument Loop/Indicator: Channel A RMG-18 Channel B RMG-7	2	1
4. Reactor Building Hydrogen Concentration Instrument Loop/Indicator: Channel A IAE-8263A/ICI-8257 Channel B IAE-8263B/ICI-8258	2	1
5. Reactor Building/RHR Sump Level Instrument Loop/Indicator: Channel A ILT-1969/ILI-1969 Channel B ILT-1970/ILI-1970	2	1
6. Reactor Coolant Outlet Temperature - T_{Hot} - Wide Range Instrument Loop/Indicator: Channel A ITE-413/ITI-413 Channel A ITE-423/ITI-423 Channel E ITE-433/ITR-413	2	1
7. Reactor Coolant Inlet Temperature - T_{Cold} - Wide Range Instrument/Loop Indicator: Channel E ITE-410/ITI-410 Channel E ITE-420/ITI-420 Channel E ITE-430/ITR-410	2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
8.	Reactor Coolant Pressure - Wide Range Instrument Loop/Indicator: Channel E IPT-402/IPI-402 Channel A IPT-403/IPI-403	2	1
9.	Pressurizer Water Level Instrument Loop/Indicator: Channel A ILT-459/ILI-459A Channel D ILT-460/ILI-460 Channel B ILT-461/ILI-461	2	1
10.	Reactor Coolant System Subcooling Margin Instrument Loop/Indicator: Channel A ITM-499A Channel B ITM-499B	2	1
11.	Reactor Vessel Level Instrument Loop/Indicator: Channel A ILT-1311/ILI-1311, ILT-1312/ILI-1312 Channel B ILT-1321/ILI-1321, ILT-1322/ILI-1322	2	1
12.	Core Exit Temperature Instrument Loop/Indicator: Channel A ITEs 2, 4, 9, 12, 13, 15, 19, 21, 22, 23, 24, 25, 26, 27, 28, 29, 31, 32, 33, 35, 39, 41, 42, 45, 46, 47 Channel B ITEs 1, 3, 5, 6, 7, 8, 10, 11, 14, 16, 17, 18, 20, 30, 34, 36, 37, 38, 40, 43, 44, 48, 49, 50, 51 (Primary display is the plant computer) (Backup displays are ITM 499 A&B)	4/core quadrant/ channel	2/core quadrant/ channel
13.	Neutron Flux Instrument Loop/Indicator: Channel 1 INI-35 Channel 2 INI-36	2	1

TABLE 3.3-10 (continued)

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
14. Steam Line Pressure Instrument Loop/Indicator: SG A IPTs-474, 475, 476/IPIs-474, 475, 476 SG B IPTs-484, 485, 486/IPIs-484, 485, 486 SG C IPTs-494, 495, 496/IPIs-494, 495, 496	2/stm. gen.	1/stm. gen.
15. Steam Generator Water Level - Wide Range Instrument Loop/Indicator: SG A ILT-477/ILI-477 SG B ILT-487/ILI-487 SG C ILT-497/ILI-497	1/stm. gen.	1/stm. gen.
16. Steam Generator Water Level - Narrow Range Instrument Loop/Indicator: SG A ILTs 474, 475, 476/ILIs 474, 475, 476 SG B ILTs 484, 485, 486/ILIs 484, 485, 486 SG C ILTs 494, 495, 496/ILIs 494, 495, 496	2/stm. gen.	1/stm. gen.
17. Emergency Feedwater Flow Instrument Loop/Indicator: Channel A SG A IFT-3561/IFI-3561 SG B IFT-3571/IFI-3571 SG C IFT-3581/IFI-3581 Channel B SG A IFT-3561A/IFI-3561B SG B IFT-3571A/IFI-3571B SG C IFT-3581A/IFI-3581B	2/stm. gen.	1/stm. gen.
18. Refueling Water Storage Tank Level Instrument Loop/Indicator: Channel A ILT-990/ILI-990 Channel B ILT-992/ILI-992	2	1

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INSTRUMENTATION

EXPLOSIVE GAS MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.9 The explosive gas monitoring instrumentation channels shown in Table 3.3-13 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.11.2.5 are not exceeded.

APPLICABILITY: As shown in Table 3.3-13

ACTION:

a. With an explosive gas monitoring instrumentation channel alarm/trip setpoint less conservative than required by the above Specification, declare the channel inoperable and take the ACTION shown in Table 3.3-13.

b. With less than the minimum number of explosive gas monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. Restore the inoperable instrumentation to operable status within 30 days and, if unsuccessful prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 to explain why this inoperability was not corrected in a timely manner.

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-13

EXPLOSIVE GAS MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
a. Oxygen Monitor	2	**	44
b. Hydrogen Monitor	1	**	42

TABLE 3.3-13 (Continued)

TABLE NOTATION

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

ACTION 38 - (Not used)

ACTION 39 - (Not used)

ACTION 40 - (Not used)

ACTION 41 - (Not used)

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

ACTION 43 - (Not used)

ACTION 44 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue provided grab samples are taken and analyzed at least once per 24 hours. With both the channels inoperable, operation may continue provided grab samples are taken and analyzed at least once per 4 hours during degassing operations and at least once per 24 hours during other operations.

TABLE 4.3-9

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM				
a. Hydrogen Monitor	D	Q(1)	M	**
b. Oxygen Monitor	D	Q(2)	M	**

TABLE 4.3-9 (Continued)

TABLE NOTATION

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

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

1. 1500 ± 30 ppm hydrogen, balance nitrogen, for the outlet hydrogen monitor and
2. 4 ± 0.1 volume percent hydrogen, balance nitrogen, for the inlet hydrogen monitor.

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

1. 75 ± 1.5 ppm oxygen, balance nitrogen, for the outlet oxygen monitor and
2. 3.5 ± 0.1 volume percent oxygen, balance nitrogen, for the inlet oxygen monitor.

INSTRUMENTATION

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.10 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

Correction letter of 12-9-91

LIMITING CONDITION FOR OPERATION

3.3.3.11 The Power Distribution Monitoring System (PDMS) shall be OPERABLE with:

a. A minimum of the following inputs from the plant available for use by the PDMS as defined in Table 3.3-14.

1. Control Bank Position
2. T_{cold}
3. Reactor Power Level
4. NIS Power Range Detector Section Signals

b. Core Exit Thermocouples (T/C) meeting the criteria:

1. At least 25% operable T/C with at least 2 T/C per quadrant, and
2. The T/C pattern has coverage of all interior fuel assemblies (no face along the baffle), within a chess knight's move, radially, from a responding, calibrated T/C, or
3. At least 25% operable T/C with at least 2 T/C per quadrant, and the installed PDMS calibration was determined within the last 31 Effective Full Power Days (EFPD).
4. The T/C temperatures used by the PDMS are calibrated via cross calibration with the loop temperature measurement RTDs, and using the T/C flow mixing factors determined during installed PDMS calibration.

c. An installed PDMS calibration satisfying the criteria:

1. The initial calibration in each operating cycle is determined using measurements from at least 75% of the incore movable detector thimbles obtained at a THERMAL POWER greater than 25% of RATED THERMAL POWER.
2. The calibration is determined using measurements from at least 50% of the incore movable detector thimbles at any time except as specified in 3.3.3.11.c.1, and
3. The calibration is determined using a minimum of 2 detector thimbles per core quadrant.

LIMITING CONDITION FOR OPERATION (Continued)

APPLICABILITY: MODE 1, above 25% RATED THERMAL POWER (RTP)

ACTION:

With any of the operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, or 3.3.3.11.c not met, either correct the deficient operability condition, or declare the PDMS inoperable and use the incore movable detector system, satisfying the OPERABILITY requirements listed in Specification 3.3.3.2, to obtain any required core power distribution measurements. Increase the measured core peaking factors using the values listed in the COLR for the PDMS inoperable condition.

The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.11.1 The operability criteria listed in 3.3.3.11.a, 3.3.3.11.b, and 3.3.3.11.c shall be verified to be satisfied prior to acceptance of the PDMS core power distribution measurement results.

4.3.3.11.2 Calibration of the PDMS is required:

a. at least once every 180 Effective Full Power Days when the minimum number and core coverage criteria as defined in 3.3.3.11.b.1 and 3.3.3.11.b.2 are satisfied, or

b. at least once every 31 Effective Full Power Days when only the minimum number criterion as defined in 3.3.3.11.b.3 is satisfied.

TABLE 3.3-14

REQUIRED PDMS PLANT INPUT INFORMATION

	PLANT INPUT INFORMATION	AVAILABLE INPUTS	MINIMUM NO. OF VALID INPUTS	APPLICABLE MODES
1.	Control Bank Position	4	4[a]	1[c]
2.	T_{cold}	3	2	1[c]
3.	Reactor Power Level	3	1[b]	1[c]
4.	NIS Power Range Excore Detector Section Signals	8	6[d]	1[c]

TABLE NOTATIONS

a. Determined from either valid Demand Position or the average of the valid individual RCCA position indications for all RCCAs in the Control Bank.

b. Determined from either the reactor THERMAL POWER derived using a valid secondary calorimetric measurement, the average NIS Power Range Detector Power, or the average RCS Loop ΔT.

c. Greater than 25% RTP.

d. Comprised of an upper and lower detector section signal per Power Range Channel; a minimum of 3 OPERABLE channels required.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 All Reactor Coolant loops shall be in operation.

APPLICABILITY: MODES 1 and 2.*

ACTION:

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

SURVEILLANCE REQUIREMENT

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

*See Special Test Exception 3.10.4.

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

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

d. Residual Heat Removal Loop A,

e. Residual Heat Removal Loop B,

APPLICABILITY: MODE 4

ACTION:

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

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

---

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

COLD SHUTDOWN - LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 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 percent of wide range indication.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

REACTOR COOLANT SYSTEM

COLD SHUTDOWN - LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5 with Reactor Coolant loops not filled.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.1 The Surveillance Requirements of Specification 4.0.5 shall be met, or; the pressurizer code safety valve shall have its lift set pressure verified under cold conditions.

Correction letter of 2-14-91

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2485 PSIG ± 3%.*

APPLICABILITY: MODES 1, 2 and 3.#

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

#Mode 3 applicability is exempted under the following conditions:

1. There has been a least 5 days of operation in MODES 5 or 6 since the reactor was last critical, and

2. All RCCAs are fully inserted with all CRDMs de-energized.

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

REACTOR COOLANT SYSTEM

3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With one or more PORV(s) inoperable and capable of being manually cycled, within 1 hour:
   1) restore the PORV(s) to OPERABLE status or
   2) close the associated block valve(s) and maintain power to the block valve;

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

b. With one PORV inoperable and not capable of being manually cycled, within 1 hour:
   1) restore the PORV to OPERABLE status or to a condition where it may be manually cycled* or
   2) close its associated block valve and remove power from the block valve;

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

c. With two PORVs inoperable and not capable of being manually cycled,
   1) within 1 hour:
      a) restore the PORVs to OPERABLE status or to a condition where they are capable of being manually cycled* or
      b) close the associated block valves and remove power from the block valves and
   2) within the next 72 hours:
      a) restore a minimum of two PORVs to OPERABLE status or
      b) restore a minimum of two PORVs to a condition where they are capable of being manually cycled*;

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

* If a PORV is inoperable but capable of being manually cycled, the associated block valve must be closed with power maintained to the block valve.

ACTION: (Continued)

d. With three PORVs inoperable and not capable of being manually cycled,
   1) within 1 hour:
      a) restore at least one PORV to OPERABLE status or to a condition where it is capable of being manually cycled*, and
      b) close and remove power from the block valves for any PORVs remaining inoperable and not capable of being manually cycled and
   2) within the next 72 hours:
      a) restore a minimum of two PORVs to OPERABLE status or
      b) restore a minimum of two PORVs to a condition where they can be manually cycled*;

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

e. With one block valve inoperable:
   1) within 1 hour:
      a) restore the block valve to OPERABLE status, or
      b) place the associated PORV in manual control and
   2) within the next 72 hours:
      a) restore the block valve to OPERABLE status or
      b) close the block valve and remove power from the block valve;

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

f. With two block valves inoperable:
   1) within 1 hour:
      a) restore the block valves to OPERABLE status, or
      b) place the associated PORVs in manual control and
   2) within 72 hours:
      a) restore at least two of the three block valves to OPERABLE status and
      b) ensure that the remaining inoperable block valve is closed and the power is removed;

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

* If a PORV is inoperable but capable of being manually cycled, the associated block valve must be closed with power maintained to the block valve.

ACTION: (Continued)

g. With three block valves inoperable:
   1) within 1 hour:
      a) restore the block valves to OPERABLE status, or
      b) place the associated PORVs in manual control and
   2) within the next 2 hours restore at least one of the three block valves to OPERABLE status and
   3) within the next 72 hours:
      a) restore at least two of the three block valves to OPERABLE status and
      b) ensure that the remaining inoperable block valve is closed and the power is removed;

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

h. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by.operating the valve through one complete cycle of full travel during MODES 3 or 4.

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

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 prior to increasing T_{avg} above 200°F.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

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

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

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

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

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

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

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

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

   4. A main steam line or feedwater line break.

* A one-time inspection interval of once per 58 months is allowed for the inspection performed immediately following refueling outage RF-12.

4.4.5.4 Acceptance Criteria

a. As used in this Specification:

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

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

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

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

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

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

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

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

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

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

4.4.5.5 Reports

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

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

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

c. Results of steam generator tube inspections which fall into Category C-3 and require prompt notification of the Commission shall be reported pursuant to 10 CFR 50.72(b)2(i) prior to resumption of plant operation. A report pursuant to 10 CFR 50.73(a)2(ii) shall be submitted to provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

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TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

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

*The other steam generator not inspected during the first inservice inspection shall be inspected. The third and subsequent inspections may be limited to one steam generator on a rotating schedule encompassing 9% of the tubes if the results of the previous inspections indicate that all steam generators are performing in a like manner. Note that under some circumstances, the operating conditions in one or more steam generators may be found to be more severe than those in other steam generators. Under such circumstances the sample sequence shall be modified to inspect the most severe conditions.

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S. G.	C–1	None	N/A	N/A	N/A	N/A
	C–2	Plug or repair defective tubes and inspect additional 2S tubes in this S. G.	C–1	None	N/A	N/A
			C–2	Plug or repair defective tubes and inspect additional 4S tubes in this S. G.	C–1	None
					C–2	Plug or repair defective tubes
					C–3	Perform action for C–3 result of first sample
			C–3	Perform action for C–3 result of first sample	N/A	N/A
	C–3	Inspect all tubes in this S. G., plug or repair defective tubes and inspect 2S tubes in each other S. G. Prompt notification to NRC pursuant to 10 CFR 50.72 (b)2(i) and 10 CFR 50.73 (a) 2(ii)	All other S. G.s are C–1	None	N/A	N/A
			Some S. G.s C–2 but no additional S. G. are C–3	Perform action for C–2 result of second sample	N/A	N/A
			Additional S. G. is C–3	Inspect all tubes in each S. G. and plug or repair defective tubes. Prompt notification to NRC pursuant to 10 CFR 50.72 (b)2(i) and 10 CFR 50.73(a)2(ii)	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

The sample size, S, will also include 3% of the total number of sleeved tubes in all 3 steam generators or all of the sleeved tubes in the generator chosen for the inspection, whichever is less.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. A reactor building atmosphere particulate radioactivity monitoring system,

b. The reactor building sump level, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

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

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

d. Reactor building cooling unit condensate flow detector-performance of CHANNEL CALIBRATION at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 150 gallons per day through any one steam generator not isolated from the Reactor Coolant System,

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

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

f. The leakage rate specified for each Reactor Coolant System Pressure Isolation Valve in Table 3.4-1 at a Reactor Coolant System pressure of 2235 ± 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

Correction letter of 12-15-2003

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

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

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

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

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

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

SURVEILLANCE REQUIREMENTS (Continued)

b. Monitoring the reactor building sump inventory at least once per 12 hours.

c. Measurement of the CONTROLLED LEAKAGE to the reactor coolant pump seals when the Reactor Coolant System pressure is 2235 ± 20 psig at least once per 31 days with the modulating valve fully open. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3 or 4.

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

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

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

a. During startup following each refueling outage.

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

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

VALVE NO.	DESCRIPTION	NOMINAL SIZE (Inches)	ALLOWABLE LEAKAGE PER VALVE (Gallons per Minute)
8993 A, B, C	SI to Hot Legs	6	3
8992 A, B, C	SI High Head to Hot Legs	2	1
8990 A, B, C	SI High Head to Hot Legs	2	1
8988 A, B	SI Low Head to Hot Legs	6	3
8997 A, B, C	Primary SI High Head to Cold Legs	2	1
8995 A, B, C	Alternate SI High Head to Cold Legs	2	1
8998 A, B, C	SI to Cold Legs	6	3
8973 A, B, C	RHR Low Head to Cold Legs	6	3
*8948 A, B, C	Accumulators to Cold Legs	12	5
*8956 A, B, C	Accumulators to Cold Legs	12	5
8701 A, B	RHR Suction from Hot Legs	12	5
8702 A, B	RHR Suction from Hot Legs	12	5
8974 A, B	RHR Low Head to Cold Legs	10	5

*See Specification 4.4.6.2.2.c

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

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

*See Specification 4.4.6.2.2.c.

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

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

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

At All Other Times:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.4-2

CHEMISTRY LIMITS

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

*Limit not applicable with $T_{avg} \leq$ 250°F.

TABLE 4.4-3

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

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

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

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

MODES 1, 2 and 3*:

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

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

MODES 1, 2, 3, 4 and 5:

a. With the specific activity of the primary coolant greater than 1.0 microcurie per gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries per gram, perform the sampling and analysis requirements of item 4a of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

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

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

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TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

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

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

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

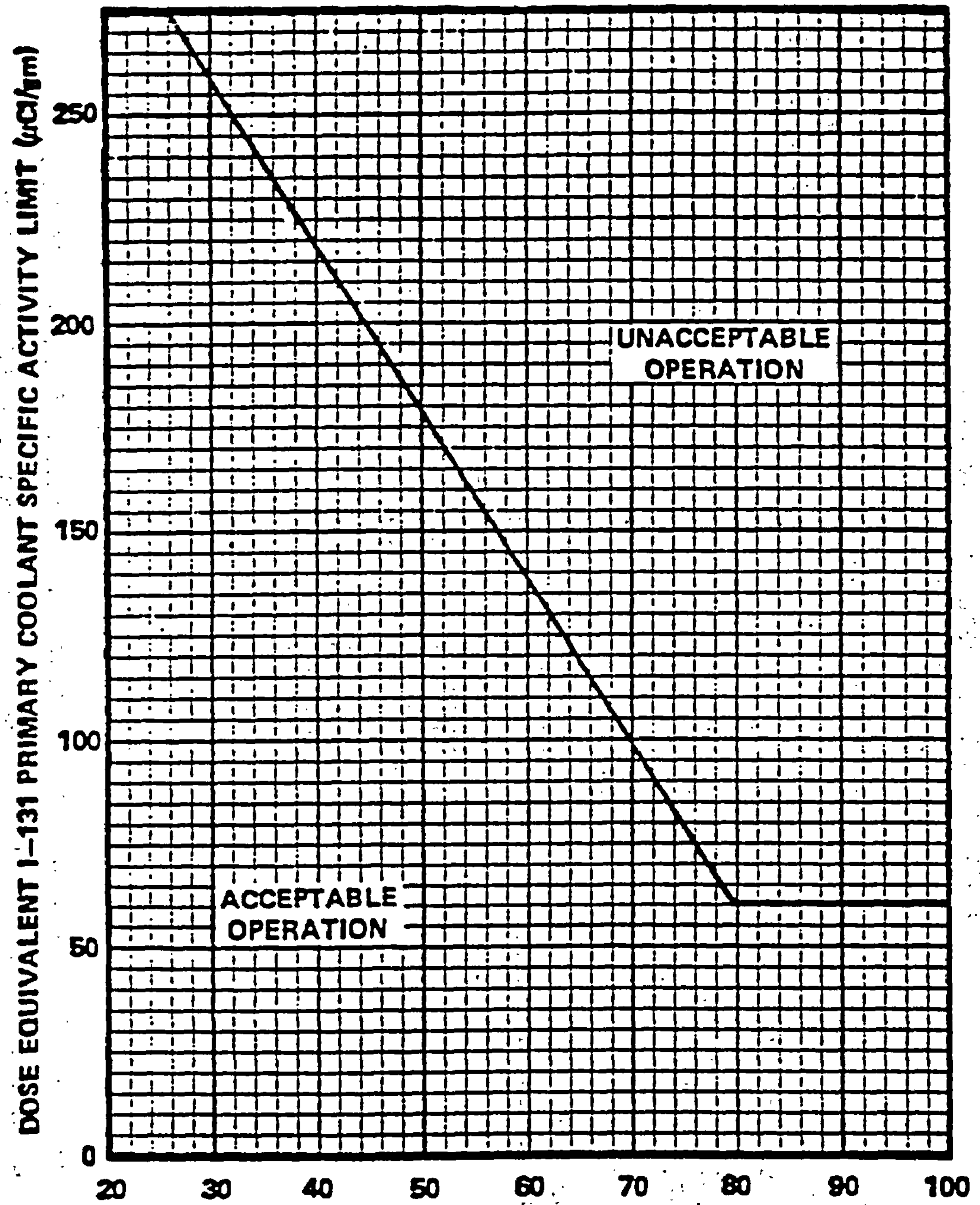

FIGURE 3.4-1

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATES C9923-1, -2

LIMITING ART VALUES AT 32 EFPY: 1/4T, 107 °F

3/4T, 94 °F

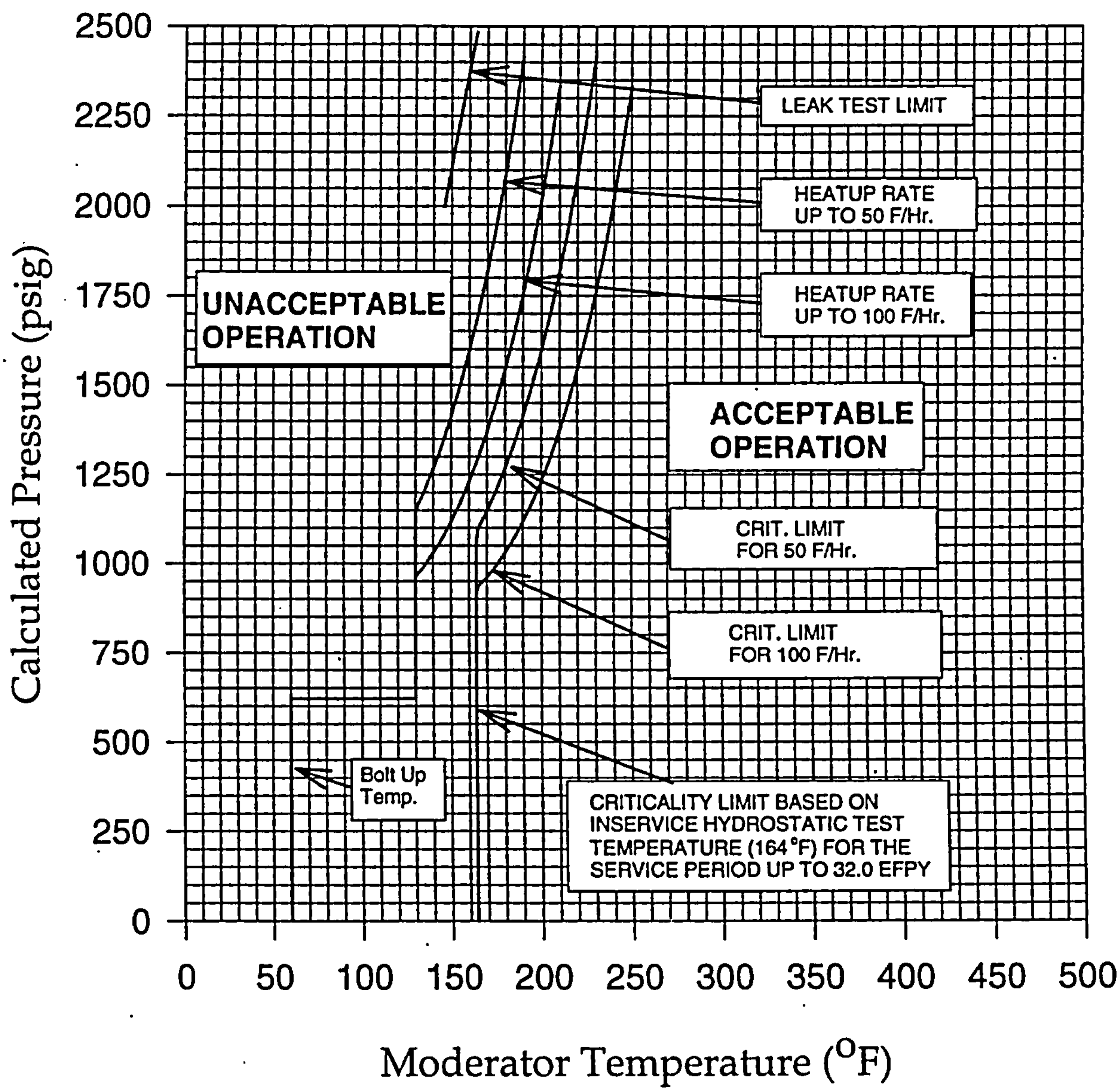

Figure 3.4-2 V. C. Summer Unit 1 Reactor Coolant System Heatup Limitations (Heatup rates up to 50 and 100 °F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: LOWER SHELL PLATES C9923-1, -2

LIMITING ART VALUES AT 32 EFPY: 1/4T, 107 °F
3/4T, 94 °F

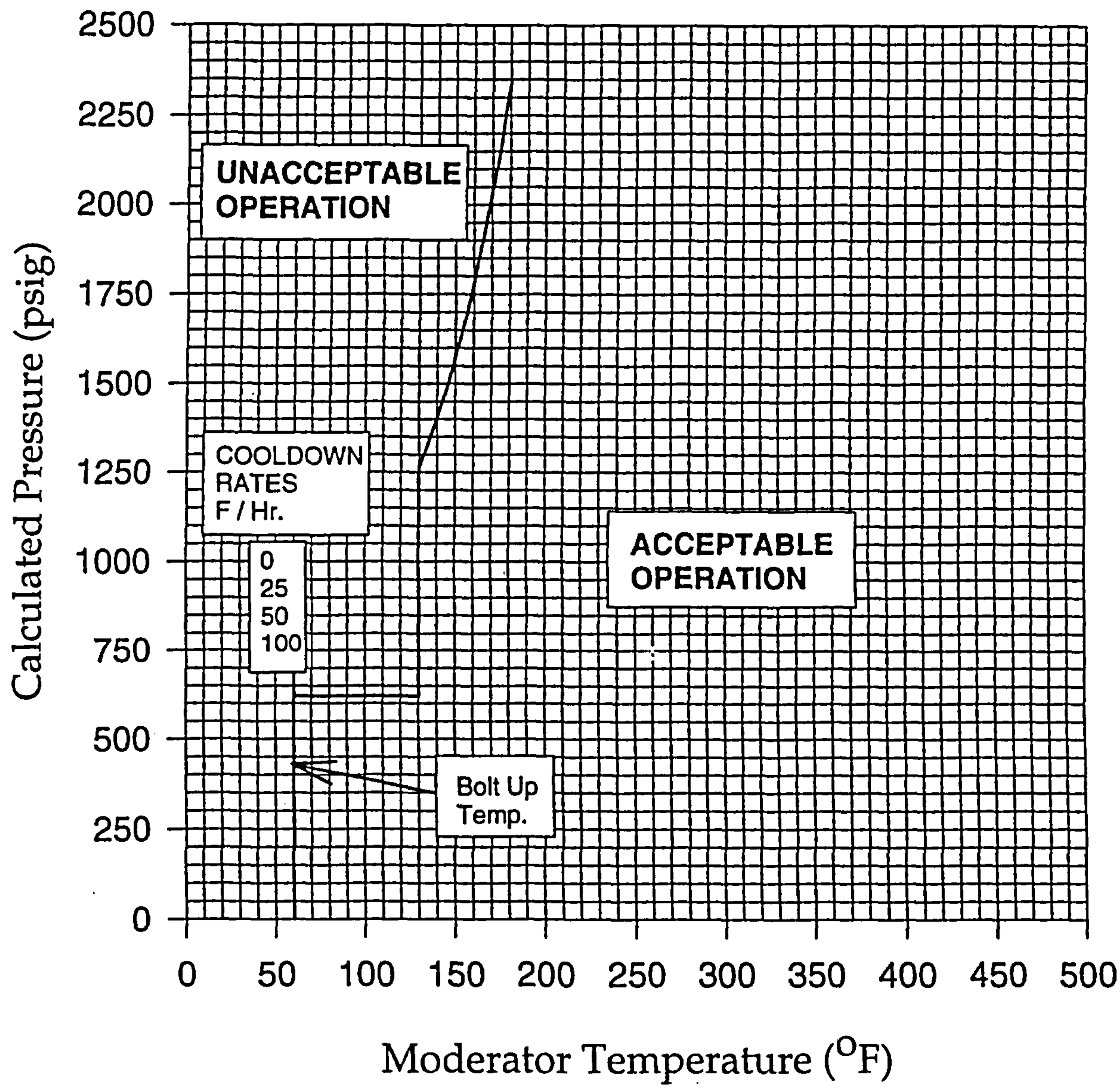

Figure 3.4-3 V. C. Summer Unit 1 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 25, 50 and 100 °F/hr) Applicable for the First 32 EFPY (Without Margins for Instrumentation Errors)

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

LIMITING CONDITION FOR OPERATION

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

a. Two RHR relief valves with:

1. A lift setting of less than or equal to 450 psig, and

2. The associated RHR relief valve isolation valves open; or

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

APPLICABILITY:

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

ACTION:

a. With one RHR relief valve inoperable, restore the inoperable valve to OPERABLE status within 72 hours or depressurize and vent the RCS through a greater than or equal to 2.7 square inch vent within the next 8 hours.

b. With both RHR relief valves inoperable, within 8 hours either:

1. Restore at least one RHR relief valve to OPERABLE status, or

2. Depressurize and vent the RCS through a greater than or equal to 2.7 square inch vent.

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

d. In the event that two or more charging pumps are capable of injecting into the RCS, immediately initiate action to ensure a maximum of one charging pump is capable of injecting into the RCS#.

e. The provisions of Specification 3.0.4 are not applicable.

Two charging pumps may be capable of injecting into the RCS while swapping pumps, ≤15 minutes.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each RHR relief valve shall be demonstrated OPERABLE by:

a. Verifying the RHR relief valve isolation valves (8701A, 8701B, 8702A, and 8702B) are open at least once per 72 hours when the RHR relief valve is being used for overpressure protection.

b. Testing pursuant to Specification 4.0.5.

c. Verification of the RHR relief valve setpoint of at least one RHR relief valve, at least once per 18 months on a rotating basis.

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

4.4.9.3.3 At least two charging pumps shall be verified incapable of injecting into the RCS at least once per 31 days, except when the reactor vessel head is removed, by verifying that the motor circuit breakers are secured in the open position.

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

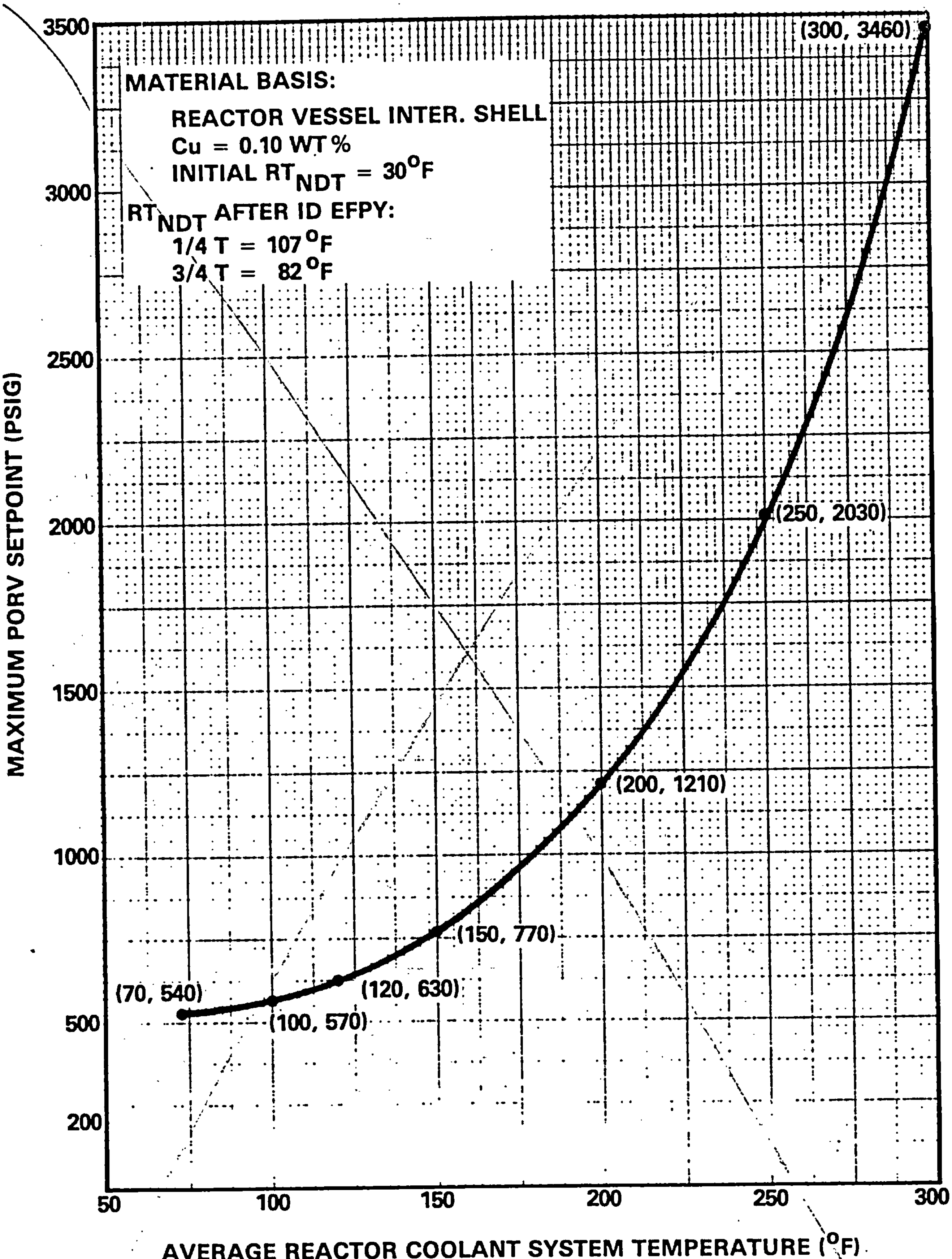

Figure 3.4-4 RCS Cold Overpressurization Protection

3/4.4.10 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES

ACTION:

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

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

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 7489 and 7685 gallons,

c. A boron concentration of between 2200 and 2500 ppm, and

d. A nitrogen cover-pressure of between 600 and 656 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

2. Verifying that each accumulator isolation valve is open.

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

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

c. At least once per 31 days when the RCS pressure is above 2000 psig by verifying that the isolation valve operator breaker opened at the motor control center and locked in the open position.

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

   1. When an actual or a simulated RCS pressure signal exceeds the P-11 (Pressurizer Pressure Block of Safety Injection) setpoint,

   2. Upon receipt of a safety injection test signal.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

* The allowable outage time for each RHR train may be extended to 7 days for the purpose of maintenance and modification. This exception may only be used one time per RHR train and is not valid after December 31, 1997.

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

	Valve Number	Valve Function	Valve Position
1.	8884	HHSI Hot Leg Injection	Closed
2.	8886	HHSI Hot Leg Injection	Closed
3.	8888A	LHSI Cold Leg Injection	Open
4.	8888B	LHSI Cold Leg Injection	Open
5.	8889	LHSI Hot Leg Injection	Closed
6.	8701A	RHR Inlet	Closed
7.	8701B	RHR Inlet	Closed
8.	8702A	RHR Inlet	Closed
9.	8702B	RHR Inlet	Closed
10.	8133A	Charging/HHSI Cross-Connect	Open
11.	8133B	Charging/HHSI Cross-Connect	Open
12.	8106	Charging Mini-Flow Header Isolation	Open

b. At least once per 31 days by:

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

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

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

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

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

d. At least once per 18 months by:

1. Verifying automatic 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 425 psig, the interlocks prevent the valves from being opened.

- effective as of its date of issuance and shall be implemented AUG 8 1997
within the tenth refueling outage scheduled to begin October 4, 1997 (see correction letter dated 8-19-97)

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

a) Centrifugal charging pump

b) Residual heat removal pump

f. By verifying each ECCS pump's developed head at the test flow point for that pump is greater than or equal to the required developed head in accordance with Specification 4.0.5.

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

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

2. At least once per 18 months.

HPSI System
Valve Number

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

a) The sum of the injection line flow rates, excluding the highest flow rate, is greater than or equal to 338 gpm, and

b) The total pump flow rate is less than or equal to 688 gpm.

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

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

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

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

A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE centrifugal charging pump,#

b. One OPERABLE residual heat removal heat exchanger,

c. One OPERABLE residual heat removal pump, and

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

APPLICABILITY: MODE 4.

ACTION:

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

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

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

A maximum of one centrifugal charging pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to 300°F.

SURVEILLANCE REQUIREMENTS

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

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

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

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

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

b. A boron concentration of between 2300 and 2500 ppm of boron, and

c. A minimum water temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

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

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

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

c. Deleted.

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

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2. Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the measured overall integrated containment leakage rate exceeding 1.0 L_a, within 1 hour initiate action to be in HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall integrated leakage rate to less than or equal to 0.75 L_a and the combined leakage rate for all penetrations subject to Type B and C tests to less than or equal to 0.60 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The containment leakage rates shall be demonstrated at the intervals derived through the Containment Leakage Rate Testing Program and shall be determined per the program criteria.

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CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each reactor building air lock shall be OPERABLE with both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one reactor building air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates in accordance with the Containment Leakage Rate Testing Program.

b. Deleted.

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

d. Deleted.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each reactor building air lock shall be demonstrated OPERABLE:

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

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

   1. At least once per 6 months#, and

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

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

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

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

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Elevation 412' - 3 locations

b. Elevation 436' - 3 locations

c. Elevation 463' - 3 locations

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

b. With the structural integrity of the containment otherwise not conforming to the requirements of Specification 4.6.1.6, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days after completion of the inspection describing the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedures, the tolerances on cracking, and the corrective actions taken.

SURVEILLANCE REQUIREMENTS

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

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

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

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.6 The structural integrity of the containment shall be OPERABLE.

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

ACTION:

If the structural integrity of the containment is found to be inoperable, restore the containment to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.6.1 The structural integrity of the containment shall be demonstrated in accordance with the Containment Inservice Inspection Program.

4.6.1.6.2 Deleted

4.6.1.6.3 In accordance with the Containment Leakage Rate Testing Program, the structural integrity of the exposed accessible interior and exterior surfaces of the containment shall be determined by a visual inspection of these surfaces and verifying that no abnormal material or structural behavior is evident.

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SURVEILLANCE REQUIREMENTS (Continued)

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CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. Closed at least once per 24 hours.

b. Sealed closed at least once per 31 days.

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

4.6.1.7.3 At least once per 30 months each containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE in accordance with the Containment Leakage Rate Testing Program.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

REACTOR BUILDING SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each reactor building spray system shall be demonstrated OPERABLE:

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

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 195 psig when tested pursuant to Specification 4.0.5.

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

   1. Verifying that each automatic valve in the flow path actuates to its correct position on each of the following test signals a Phase 'A', Reactor Building Spray Actuation, and Containment Sump Recirculation.

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

d. At least once per 10 years by performing an air or smoke or equivalent flow test through each spray header and verifying each spray nozzle is unobstructed.

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with:

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

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

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

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

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

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

REACTOR BUILDING COOLING SYSTEM

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days by:

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

b. At least once per 18 months by:

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

3. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a simulated SI test signal or on an ESFLS, as applicable.

4. At least once per 18 months, by verifying that each service water system booster pump starts automatically on a safety injection signal.

CONTAINMENT SYSTEMS

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a and C.5.c of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 60,270 ACFM ± 10%.

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

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

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

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

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

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

CONTAINMENT SYSTEMS

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

The provisions of Specification 3.0.4 do not apply.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each containment isolation valve shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair, or replacement work is performed on the valve or its associated actuator, control, or power circuit by performance of a cycling test and verification of isolation time.

4.6.4.2 Each containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE AT LEAST ONCE PER 18 MONTHS BY:

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

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

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

*Locked or sealed closed valves may be opened on an intermittent basis under administrative control.

SURVEILLANCE REQUIREMENTS (continued)

4.6.4.3 The isolation time of each power operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

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Pages 3/4 6-19, 3/4 6-20, 3/4 6-20a, 3/4 6-20b,and 3/4 6-20c deleted

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ELECTRIC HYDROGEN RECOMBINERS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.5.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

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

b. At least once per 18 months by:

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

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

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

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With 3 reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

*All valves tested must have "as-left" lift setpoints that are within ±1% of the Lift Setting value listed in Table 3.7-2.

TABLE 3.7-1

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

Maximum Number of Inoperable Safety Valves on Each Operating Steam Generator	Maximum Allowable Power Range Neutron Flux High Setpoint (Percent of RATED THERMAL POWER)
1(1)	58(1)
2	41
3	24

Notes: 1. With one inoperable safety valve in only one operating steam generator, the Maximum Allowable Power Range Neutron Flux High Setpoint may be increased to 81% of RATED THERMAL POWER provided the predicted Moderator Temperature Coefficient is negative (< 0 pcm/°F) at hot zero power assuming all rods out and no xenon.

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

S/G A	S/G B	S/G C		Lift Setting *		Orifice Size
XVS-2806A	XVS-2806F	XVS-2806K		1176 psig ± 1%		4.515 In dia/16 sq in
XVS-2806B	XVS-2806G	XVS-2806L		1190 psig ± 3%		4.515 In dia/16 sq in
XVS-2806C	XVS-2806H	XVS-2806M		1205 psig ± 3%		4.515 In dia/16 sq in
XVS-2806D	XVS-2806I	XVS-2806N		1220 psig ± 3%		4.515 In dia/16 sq in
XVS-2806E	XVS-2806J	XVS-2806P		1235 psig ± 3%		4.515 In dia/16 sq in

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

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PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3 7.1.2 At least three independent steam generator emergency feedwater pumps and flow paths shall be OPERABLE with:

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

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor driven pump develops a discharge pressure of greater than or equal to 1600 psig at greater than or equal to 90 gpm flow.

2. Verifying that the steam turbine driven pump develops a discharge pressure of greater than or equal to 1330 psig at a flow of greater than or equal to 97 gpm when the secondary steam supply pressure is greater than 865 psig. The provisions of Specification 4.0.4 are not applicable.

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

Correction letter of 9-1-93

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

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

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 172,700 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE at least once per 12 hours by verifying service water system pressure whenever the service water system is the supply source for the emergency feedwater pumps.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a contained volume of at least 179,850 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours, or

b. Demonstrate the OPERABILITY of the service water system as a backup supply to the emergency feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the emergency feedwater pumps.

4.7.1.3.2 The service water system shall be demonstrated OPERABLE at least once per 12 hours by verifying service water system pressure whenever the service water system is the supply source for the emergency feedwater pumps.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131.

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

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microcuries/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours.
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours;

Otherwise, reduce power to less than or equal to 5 percent of RATED THERMAL POWER within the next 2 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 7 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEMS

FEEDWATER ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each feedwater isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

MODE 1 With one feedwater isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 72 hours;

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3 With one feedwater isolation valve inoperable, subsequent operation in MODES 2 or 3 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each feedwater isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of the primary coolant and the steam generator shells shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

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

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

SURVEILLANCE REQUIREMENTS

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

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

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

3/4.7.4 SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one service water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 At least two service water loops shall be demonstrated OPERABLE:

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

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5 The service water pond (ultimate heat sink) shall be OPERABLE with:

a. A minimum water level at or above elevation 416.5 Mean Sea Level, USGS datum, and

b. A water temperature of less than or equal to 90.5°F at the discharge of the service water pumps.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

3/4.7.6 CONTROL ROOM NORMAL AND EMERGENCY AIR HANDLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 Two independent control room normal and emergency air handling systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4:

With one control room normal and emergency air handling system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room normal and emergency air handling system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency air cleanup system in the recirculation mode.

b. With both control room emergency air cleanup systems inoperable, or with the OPERABLE control room emergency air cleanup system, required to be in the recirculation mode by ACTION (a), not capable of being powered by an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.6 Each control room normal and emergency air handling system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 85°F.

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

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

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 21,270 SCFM ± 10%.

2. Verifying, within 31 days after removal, that a laboratory analysis of a representative charcoal sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 70% and 30°C with a methyl iodide penetration of <2.5%.

3. Verifying a system flow rate of 21,270 SCFM ± 10% during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal, that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 70% and 30°C with a methyl iodide penetration of <2.5%.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA and roughing filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 21,270 SCFM ± 10%.

2. Verifying that on a simulated SI or high radiation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that on a simulated SI or high radiation test signal the system starts the normal and emergency air handling systems which pressurize the control room to a positive pressure of greater than or equal to 1/8 inch W.G. relative to the outside atmosphere and maintains the 1/8 inch W.G. positive pressure with a maximum of 1000 SCFM of outside air during system operation.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 21,270 SCFM ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 21,270 SCFM ± 10%.

3/4.7.7 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.7 All snubbers on systems required for safe shutdown/accident mitigation shall be OPERABLE. This includes safety and non-safety related snubbers on systems used to protect the code boundary and to ensure the structural integrity of these systems under dynamic loads.

APPLICABILITY: MODES 1, 2, 3 and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.7.g on the attached component or declare the attached system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.7 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Inspection Types

As used in this specification, type of snubber shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as accessible or inaccessible during power operation. Each of these categories may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each category of snubber shall be determined based on the criteria provided in Table 4.7-2, and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment No. 103.

c. Refueling Outage Inspections

Each refueling outage an inspection shall be performed of all the snubbers defined in Section 3.7.7 attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using at least one of the following:

(i) manually induced snubber movement; (ii) evaluation of in-place snubber piston setting; or (iii) strcking the mechanical snubber through its full range of travel.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY and (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubbers to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (i) the cause for being classified as unacceptable is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (ii) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.7.f. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and shall not be determined OPERABLE via functional testing unless the test is started with the piston in the as found setting, extending the piston rod in the tension mode direction. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable and may be reclassified as acceptable for determining the next inspection interval provided that criterion (i) and (ii) above are met. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements of 3.7.7 shall be met.

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter, a representative sample of either: (1) At least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.7.f, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested, or (2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1, "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.7.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Accept" region, testing of that type of snubber may be terminated.

e. Functional Tests (Continued)

When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or all the snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers of each type. The representative sample shall be weighted to include more snubbers from severe service areas such as near heavy equipment. Snubbers placed in the same location as snubbers which failed the previous functional test shall be included in the next test lot if the failure analysis shows that failure was due to location.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression, except that inertia dependent, acceleration limiting mechanical snubbers, may be tested to verify only that activation takes place in both directions of travel.

2. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range.

3. Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both direction of travel.

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

5. Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

g. Functional Test Failu e Analysis (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.7.e for snubbers not meeting the functional test acceptance criteria.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals service life is not exceeded between surveillance inspections. The maximum expected service life for the various seals, scal materials, and applications shall be determined and established based on engineering information and the seals shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

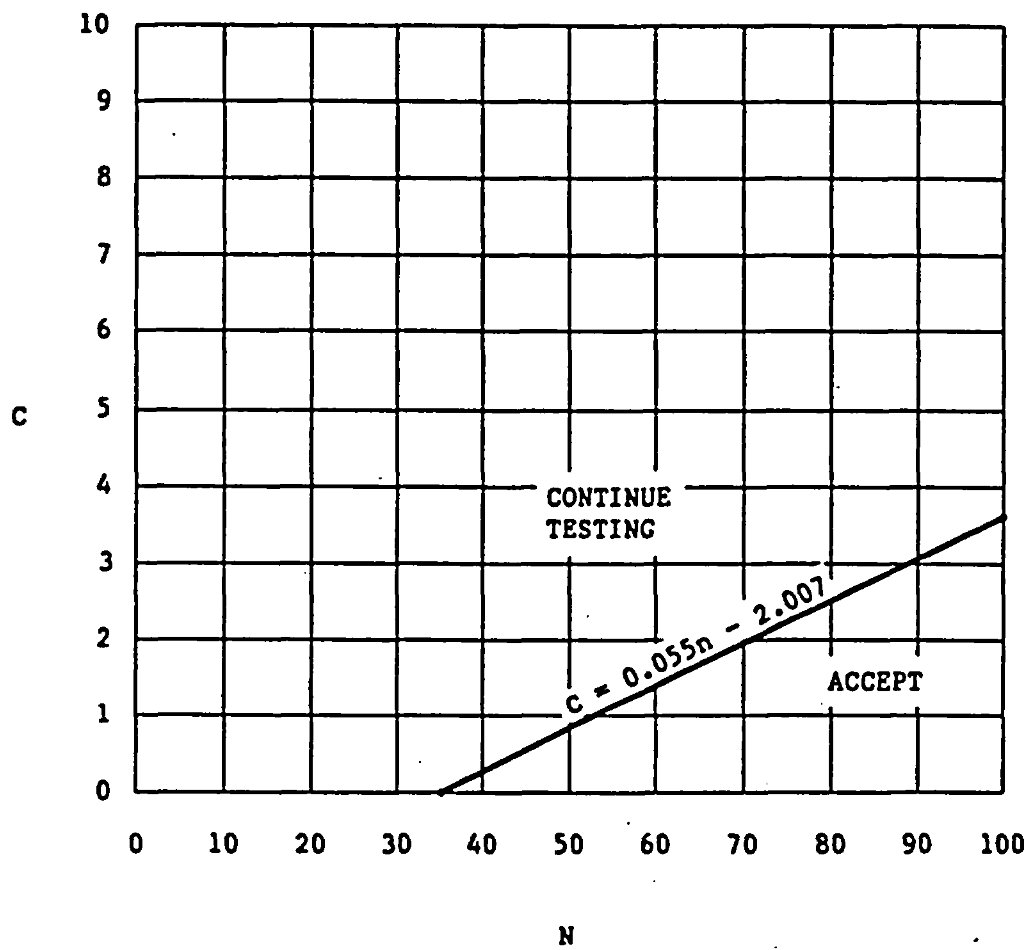

FIGURE 4.7-1 SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

TABLE NOTATION

(1) The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that internal. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

(2) Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that includes a fractional value of unacceptable snubbers as determined by interpolation.

(3) If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

(4) If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

(5) If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ration of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

(6) The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

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3/4.7.8 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

   1. Decontaminate and repair the sealed source, or

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

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

SURVEILLANCE REQUIREMENTS

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

a. The licensee, or

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

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

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

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

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

   2. In any form other than gas.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source.

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

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Pages 3/4 7-26 through 3/4 7-36 have been deleted.

3/4.7.9 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.9 The temperature of each area shown in Table 3.7-7 shall be maintained below the limits indicated in Table 3.7-7.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) shown in Table 3.7-7:

a. For more than eight hours, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days providing a record of the amount by which and the cumulative time the temperature in the affected area exceeded its limit and an analysis to demonstrate the continued OPERABILITY of the affected equipment.

b. By more than 30°F, in addition to the Special Report required above, within 4 hours either restore the area to below its temperature limit or declare the equipment in the affected area inoperable.

SURVEILLANCE REQUIREMENTS

4.7.9 The temperature in each of the areas of Table 3.7-7 shall be determined to be within its limit at least once per 12 hours.

TABLE 3.7-7

AREA TEMPERATURE MONITORING

	AREA	TEMPERATURE LIMIT (°F)
1.	Charging-SI Pump Room #1	102
2.	Charging-SI Pump Room #2 (swing)	102
3.	Charging-SI Pump Room #3	102
4.	RHR-Spray Pump Room #1	102
5.	RHR-Spray Pump Room #2	102
6.	MCC 1DA2Y Room	102
7.	Switchgear 1DB1 and MCC 1DB2Y Room	102
8.	Switchgear 1DA Room	102
9.	Switchgear 1DB Room	102
10.	Battery 1A Room	88
11.	Battery 1B Room	88
12.	Charger 1A Room	102
13.	Charger 1B Room	102
14.	Charger 1A/1B Room	102
15.	Relay Room	83
16.	Component Cooling pump "A" Speed Switch Room	102
17.	Component Cooling pump "B" Speed Switch Room	102
18.	Component Cooling Pump "C" Speed Switch Room	102
19.	Evacuation Panel "A" Room	83
20.	Evacuation Panel "B" Room	83
21.	Service Water Booster Pumps Area	102
22.	Emergency Feedwater Pumps Area	102
23.	Diesel Generator 1A Room	120
24.	Diesel Generator 1B Room	120
25.	Service Water Pump/Screen Room	118
26.	Service Water Switchgear Room "A"	102
27.	Service Water Switchgear Room "B"	102
28.	Service Water Switchgear Room "C"	102
29.	Diesel Generator Exciter Cabinet Room "A"	102
30.	Diesel Generator Exciter Cabinet Room "B"	102

3/4.7.10 WATER LEVEL-SPENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.7.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

SURVEILLANCE REQUIREMENTS

4.7.10 The water level in the spent fuel pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the spent fuel pool.

3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.11 Two independent spent fuel pool ventilation sub-systems shall be OPERABLE with at least one sub-system in operation.

APPLICABILITY: Whenever irradiated fuel is being moved in the spent fuel pool and during crane operation with loads over the pool.

ACTION:

a. With one spent fuel pool ventilation sub-system inoperable, fuel movement within the spent fuel pool or crane operation with loads over the spent fuel pool may proceed provided the OPERABLE spent fuel pool ventilation sub-system is capable of being powered from an OPERABLE emergency power source and is in operation and discharging through at least one train of HEPA filters and charcoal adsorbers.

b. With no spent fuel pool ventilation sub-system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.11 The above required spent fuel pool ventilation sub-systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that each sub-system operates for at least 15 minutes.

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

   1. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 30,000 ACFM ± 10%.

2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%.

3. Verifying a system flow rate of 30,000 ACFM ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. Prior to the movement of fuel or crane operation with loads over the pool by verifying that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of ASTM D3803-1989, at a relative humidity of 95% and 30°C with a methyl iodide penetration of <2.5%. Subsequent to each initial analysis (which must be completed prior to fuel movement or crane operation with loads over the pool), during the period of time in which there is to be fuel or crane movement with loads over the pool, verify charcoal adsorber operation every 720 hours by obtaining and analyzing a sample as described above. These subsequent analyses are to be completed within thirty-one (31) days of sample removal.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA and roughing filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 30,000 ACFM ± 10%.

2. Verifying that on a loss of offsite power test signal, the system automatically starts.

3. Verifying that the system maintains the spent fuel pool area at a negative pressure greater than or equal to 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 ACFM ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 30,000 ACFM ± 10%.

PLANT SYSTEMS

3/4.7.12 SPENT FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.7.12 The combination of initial enrichment and cumulative burnup for spent fuel assemblies stored in Region 2 shall be within the acceptable domain of Figure 3.7-1.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel pool.

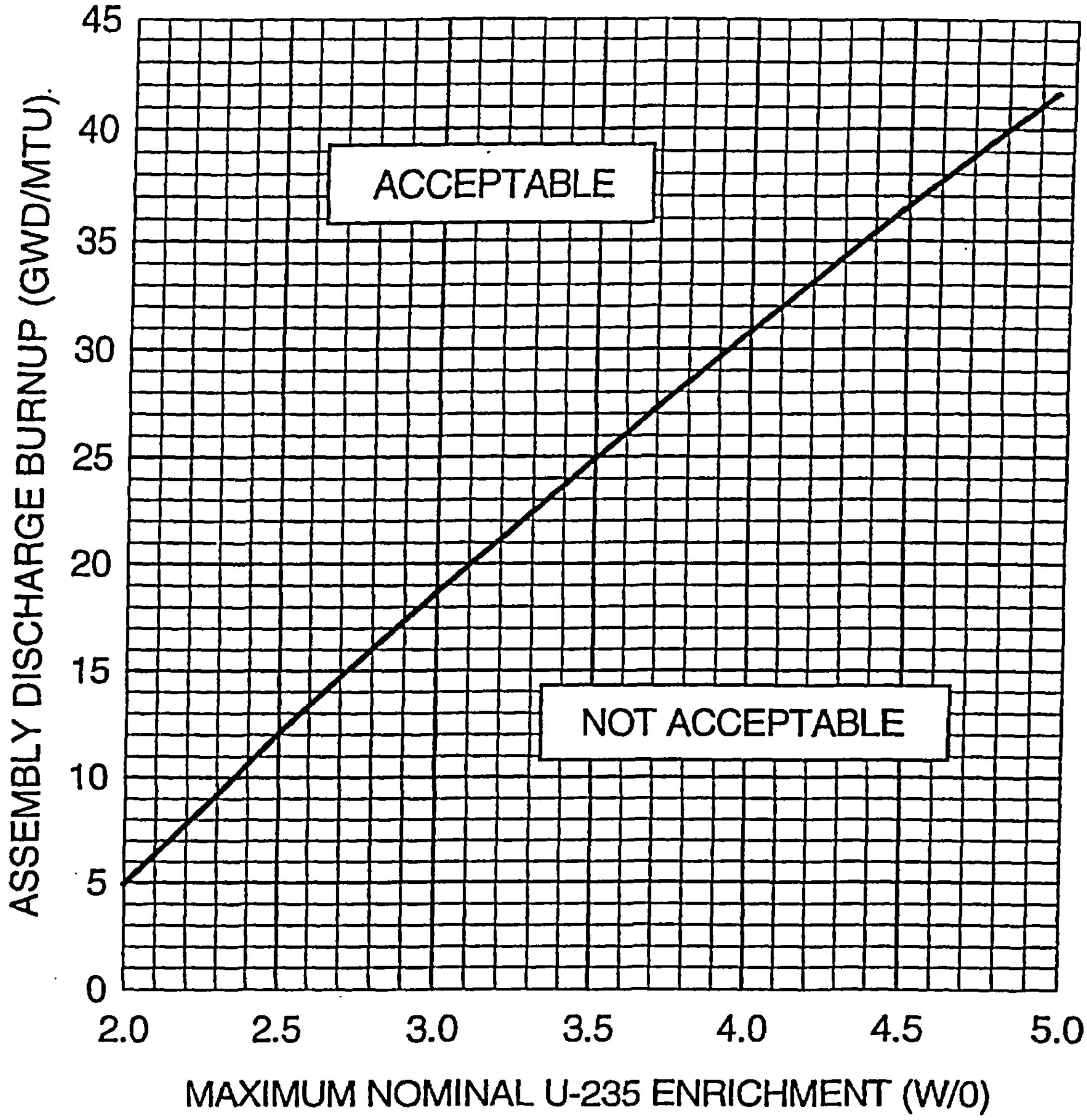
ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to Region 1. Until these requirements of the above specification are satisfied, boron concentration of the spent fuel pool shall be verified to be greater than or equal to 2000 ppm at least once per 8 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.12 The burnup of each fuel assembly stored in Region 2 shall be ascertained by careful analysis of its burnup history prior to storage in Region 2. A complete record of such analysis shall be kept for the time period that the fuel assembly remains in Region 2 of the spent fuel pool.



Notes:

1. Fuel assemblies with enrichments less than 2.0 W/0 must meet the burn-up requirements of 2.0 W/0 U-235 assemblies.
2. Use of the following polynomial fit is acceptable, where E = Enrichment (W/0):

   Assembly Discharge Burnup = $0.1246\ E^3 - 1.91\ E^2 + 20.9205\ E - 30.2482$

FIGURE 3.7-1 REQUIRED FUEL ASSEMBLY BURN-UP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.7.13 The boron concentration in the spent fuel pool, the fuel transfer canal, and the cask loading pit shall be maintained at a boron concentration greater than or equal to 500 ppm.

APPLICABILITY: Whenever new or irradiated fuel is being moved (non-refueling movement) in the spent fuel pool, fuel transfer canal, or cask loading pit.

ACTION:

With the requirements of the above not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel pool, the fuel transfer canal, and the cask loading pit until the boron concentration in the area where fuel is being moved shall be verified greater than or equal to 500 ppm.

SURVEILLANCE REQUIREMENTS

4.7.13 The boron concentration of the spent fuel pool, fuel transfer canal, or cask loading pit shall be determined by chemical analysis at least once per 72 hours when moving new or irradiated fuel in the spent fuel pool, transfer canal, or cask loading pit.

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent Emergency Diesel Generators (EDG), each with:

  1. A separate day fuel tank containing a minimum volume of 360 gallons of fuel,

  2. A separate fuel storage system containing a minimum volume of 48,500 gallons of fuel, and

  3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable:

  1. Demonstrate the OPERABILITY of the remaining offsite A.C. sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter, and

  2. If either EDG has not been successfully tested within the past 24 hours, demonstrate its OPERABILITY by performing Surveillance Requirement 4.8.1.1.2.a.3 separately for each such EDG within 24 hours unless the diesel is already operating, and

  3. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one EDG of 3.8.1.1.b inoperable:

  1. Demonstrate the OPERABILITY of the A.C. offsite sources by performing Surveillance Requirement 4.8.1.1.1 within 1 hour and at least once per 8 hours thereafter, and

  2. *If the EDG became inoperable due to any cause other than preplanned preventive maintenance or testing:

     a) determine the OPERABLE EDG is not inoperable due to a common cause failure within 24 hours, or

     b) demonstrate the OPERABILITY of the remaining EDG by performing Surveillance Requirement 4.8.1.1.2.a.3 within 24 hours,

     and

* Completion of Action b.2 is required regardless of when the inoperable EDG is restored to OPERABILITY.

ACTION: (Continued)

3. Within 2 hours, verify that required systems, subsystems, trains, components and devices that depend on the remaining EDG as a source of emergency power are also OPERABLE and in MODE 1, 2, or 3, that the Turbine Driven Emergency Feed Pump is OPERABLE. If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. Restore the EDG to OPERABLE status within 72 hours or be in at least HOT STANDY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one offsite circuit and one EDG inoperable:

1. Demonstrate the OPERABILITY of the remaining offsite A.C. source by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter, and

2. *If the EDG became inoperable due to any cause other than preplanned preventative maintenance or testing:
   a) determine the OPERABLE EDG is not inoperable due to a common cause failure within 8 hours, or
   b) demonstrate the OPERABILITY of the remaining EDG by performing Surveillance Requirement 4.8.1.1.2.a.3 within 8 hours,

   and

3. Within 2 hours, verify that required systems, subsystems, trains, components and devices that depend on the remaining EDG as a source of emergency power are also OPERABLE and in MODE 1, 2, or 3, that the Turbine Driven Emergency Feed Pump is OPERABLE. If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

4. Restore one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours, and

5. Restore the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 Action Statement a. or b., as appropriate, with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable A.C. power source.

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* Completion of Action c.2 is required regardless of when the inoperable EDG is restored to OPERABILITY.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With two of the required offsite A. C. circuits inoperable:

1. Demonstrate the OPERABILITY of the two EDG's by sequentially performing Surveillance Requirement 4.8.1.1.2.a.3 on both within 8 hours, unless the EDG's are already operating, and

2. Restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours.

3. Following restoration of one offsite source, follow Action Statement a. with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit.

e. With two of the above required EDG's inoperable:

1. Demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1 within one hour and at least once per 8 hours thereafter, and

2. Restore one of the inoperable EDG's to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

3. Following restoration of one EDG, follow Action Statement b. with the time requirement of that Action Statement based on the time of initial loss of the remaining inoperable diesel generator.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indication of power availability for each Class 1E bus and its preferred offsite power source.

4.8.1.1.2 Each EDG shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day tank and fuel storage tank.

2. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank.

3. Verifying the diesel generator can start* and accelerate to synchronous speed (504 rpm) with generator voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz.

4. Verifying the generator is synchronized, gradually loaded* to an indicated 4150-4250 kW** and operates for at least 60 minutes.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by removing accumulated water from the day tank.

c. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks.

d. By sampling new fuel oil based on the applicable ASTM standard prior to addition to storage tanks and:

1. By verifying based on the tests specified in the applicable ASTM standard prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate, or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89, or an API gravity of greater than or equal to 27 degrees but less than or equal to 39 degrees;

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes (alternatively, Saybolt viscosity, SUS at 100°F of greater than or equal to 32.6, but less than or equal to 40.1), if gravity was not determined by comparison with the supplier's certification;

* This test shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube and warmup procedures, and as applicable regarding loading recommendations.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

c) A flash point equal to or greater than 125°F; and

d) A clear and bright appearance when tested based on the applicable ASTM standard.

2. By verifying within 30 days of obtaining the sample that the specified properties are met when tested based on the applicable ASTM standard.

e. At least once every 31 days by obtaining a sample of fuel oil based on the applicable ASTM standard, and verifying that total contamination is less than 10 mg/liter when checked based on the applicable ASTM standard.

f. At least once per 184 days by:

1. Verify each EDG starts from standby conditions and:
   a) In less than or equal to 10 seconds, achieves a voltage greater than 6480 volts (7200 - 720 volts) and a frequency greater than 58.8 Hz (60 - 1.2 Hz).
   b) Achieve a steady state voltage greater than 6480 volts but less than 7920 volts and a steady state frequency greater than 58.8 Hz but less than 61.2 Hz.

   The EDG shall be started for this test by using one of the following signals:
   a) Simulated loss of offsite power by itself.
   b) Simulated loss of offsite power in conjunction with an ESF actuation test signal.
   c) An ESF actuation test signal by itself.
   d) Simulated degraded offsite power by itself.
   e) Manual.

2. The generator shall be manually synchronized, loaded to an indicated 4150-4250 kW** in less than or equal to 60 seconds, and operate for at least 60 minutes.

g. At least once every 18 months by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying that on rejection of a load of greater than or equal to 729 kW, the voltage and frequency are maintained at 7200 ± 720 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 4250 kW without tripping. The generator voltage shall not exceed 7920 volts during and following the load rejection.

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

4. Simulating a loss of offsite power by itself, and:

    a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

    b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

5. Verifying that on an ESF actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. Verify that the EDG starts from standby conditions and in less than or equal to 10 seconds, achieves a voltage greater than 6480 volts and a frequency greater than 58.8 Hz. After steady state operation is obtained, the EDG shall be verified to have a voltage greater than 6480 volts but less than 7920 volts and a frequency greater than 58.8 Hz but less than 61.2 Hz. After 5 minutes of standby operation verify that on a simulated loss of offsite power:

    a) the loads are shed from the emergency busses,

    b) the diesel generator does not connect to the bus for at least 5 seconds, and

    c) that subsequent loading of the diesel generator is in accordance with design requirements.

6. Simulating a loss of offsite power in conjunction with an ESF actuation test signal, and

    a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

    b) Verifying the EDG starts in the emergency mode, energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes and maintains the steady state voltage and frequency at 7200 ± 720 volts and 60 ± 1.2 Hz.

    c) Verifying that all EDG trips, except engine overspeed, generator differential and low lube oil pressure are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

7. Verifying the EDG operates for at least 24 hours:

   a) The EDG shall be loaded to the continuous rating (4150-4250 kW**) for the time required to reach engine temperature equilibrium, at which time the EDG shall be loaded to an indicated target value of 4676 kW (between 4600-4700 kW**) and maintained for 2 hours.

   b) During the remaining 22 hours of this test, the EDG shall be loaded to an indicated 4150-4250 kW**.

   c) During this test the steady state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

8. Verifying that the auto-connected loads to each EDG do not exceed the 2000 hour rating of 4548 kW.

9. Verifying the EDG's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode, connected to its bus, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day tank of each diesel via the installed cross connection lines.

12. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

13. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:

   a) Barring Device
   b) Remote-Local-Maintenance Switch

14. Verifying that within 5 minutes of operating the diesel generator for at least 1 hour at a load of 4150-4250 kW** the diesel starts on the auto-start signal (Loss of Off-Site Power signal), energizes the emergency busses with permanently connected loads

** This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band shall not invalidate the test.

SURVEILLANCE REQUIREMENTS (Continued)

within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer, and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization of these loads, the steady-state voltage and frequency shall be maintained at 7200 ± 720 volts and 60 ± 1.2 Hz.

h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 504 rpm in less than or equal to 10 seconds.

i. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypochlorite solution or its equivalent, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III subsection ND of the ASME Code in accordance with Specification 4.0.5.

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(Table 4.8-1 was deleted)

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator* with:

1. A day fuel tank containing a minimum volume of 360 gallons of fuel,

2. A fuel storage system containing a minimum volume of 42,500 gallons of fuel, and

3. A fuel transfer pump,

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS positive reactivity changes, movement of irradiated fuel, or crane operation with loads over the fuel storage pool. In addition, when in MODE 5 with the Reactor Coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 (with the exception of 4.8.1.1.2.a.4).

* ESF load sequencer may be deenergized in Modes 5 and 6 provided that the loss of voltage and degraded voltage relays are disabled.

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery bank No. 1A and its associated full capacity charger.

b. 125-volt Battery bank No. 1B and its associated full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129 volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

   1. The parameters in Table 4.8-2 meet the Category B limits,

   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

   3. The average electrolyte temperature of 10 of the connected cells is $\geq$ 60°F.

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

   1. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration,

   2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

   4. The battery charger will supply at least 300 amperes at 132 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1.d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A[1]	CATEGORY B[2]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	≥ 1.200[b]	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195[b]

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than (2) amps when on charge.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt battery bank and its associated full capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; and initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible.

b. With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with tie breakers open between redundant busses:

a. Train A A.C. Emergency Busses consisting of:
   1. 7200 volt Emergency Busses # 1DA and 1EA.
   2. 480 volt Emergency Busses # 1DA1, 1DA2 and 1EA1.

b. Train B A.C. Emergency Busses consisting of:
   1. 7200 volt Emergency Busses # 1DB and 1EB.
   2. 480 volt Emergency Busses # 1DB1, 1DB2, and 1EB1.

c. 120 volt A.C. Vital Busses # 5902 and 5901 energized from an associated inverter connected to D.C. Bus # 1HA*.

d. 120 volt A.C. Vital Busses # 5904 and 5903 energized from an associated inverter connected to D.C. Bus # 1HB*.

e. 120 volt A.C. Vital Bus #5907 energized.

f. 120 volt A.C. Vital Bus #5908 energized.

g. 125 volt D.C. Bus 1HA energized from Battery Bank #1A.

h. 125 volt D.C. Bus 1HB energized from Battery Bank #1B.

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

ACTION:

a. With one of the required trains of A.C. Emergency busses not fully energized, re-energize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Vital Bus not energized, re-energize the A.C. Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one of A.C. Vital Busses #5901, 5902, 5903, or 5904 either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus re-energize the A.C. Vital Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

*The inverters may be disconnected from their D.C. Bus for up to 24 hours as necessary for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. Bus.

ACTION: (Continued)

d. With one D.C. bus not energized from its associated Battery Bank, re-energize the D.C. bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. Emergency Busses consisting of two 7200 volt and three 480 volt A.C. Emergency Busses.

b. Three 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. One 125 volt D.C. Bus energized from its associated battery bank.

APPLICABITLITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, and initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.1 For each containment penetration provided with a penetration conductor overcurrent protective device(s), each device(s) shall be operable.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the above required containment penetration conductor overcurrent protective device(s) inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuit(s) by tripping the associated backup circuit breaker or racking out or removing the inoperable circuit breaker within 72 hours, declare the affected system or component inoperable, and verify the backup circuit breaker to be tripped or the inoperable circuit breaker racked out, or removed, at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, their inoperable circuit breakers racked out, or removed, or

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

SURVEILLANCE REQUIREMENTS

4.8.4.1 Protective devices required to be operable as containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE.

a. At least once per 18 months:

1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(b) An integrated system functional test which includes simulat automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breakers nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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ELECTRICAL POWER SYSTEMS

CIRCUIT PROTECTION DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4.3 Circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that if faulted could cause failure in both adjacent, redundant Class 1E cables shall be OPERABLE.

APPLICABILITY: All modes

ACTION:

a. With one or more of the above required non-Class 1E circuit breaker(s) inoperable, within 72 hours, either:

   1. Restore the circuit breaker(s) to OPERABLE status; or
   2. De-energize the circuit breaker(s); or
   3. Establish a one (1) hour roving fire watch for those areas in which redundant systems or components could be damaged.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.4.3 The above required circuit breakers shall be demonstrated OPERABLE.

a. At least once per eighteen (18) months:

   1. By verifying that the medium voltage (7.2 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing the following:

      (a) A CHANNEL CALIBRATION of the associated protective relays, and

      (b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed.

      (c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. By selecting and functionally testing a representative sample of at least ten percent (10%) of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. Testing of these circuit breakers shall consist of injecting a current in excess of the breaker's nominal setpoint and measuring the response time. The measured response time will be compared to the manufacturer's data to insure that it is less than or equal to a value specified by the manufacturer. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least ten percent (10%) of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

b. At least once per sixty (60) months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 2000 ppm.

APPLICABILITY: MODE 6* with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 2000 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 The following valves shall be verified locked closed** at least once per 72 hours: 8430, 8454, 8441 and 8439.

---

* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

**Valves may be opened under administrative control to add borated makeup.

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be OPERABLE each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable or not operating, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the reactor coolant system at least once per 12 hours.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. An ANALOG CHANNEL OPERATIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. An ANALOG CHANNEL OPERATIONAL TEST at least once per 7 days.

REFUELING OPERATIONS

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical a period of time within the acceptable domain of Figure 3.9-1, but not less than 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

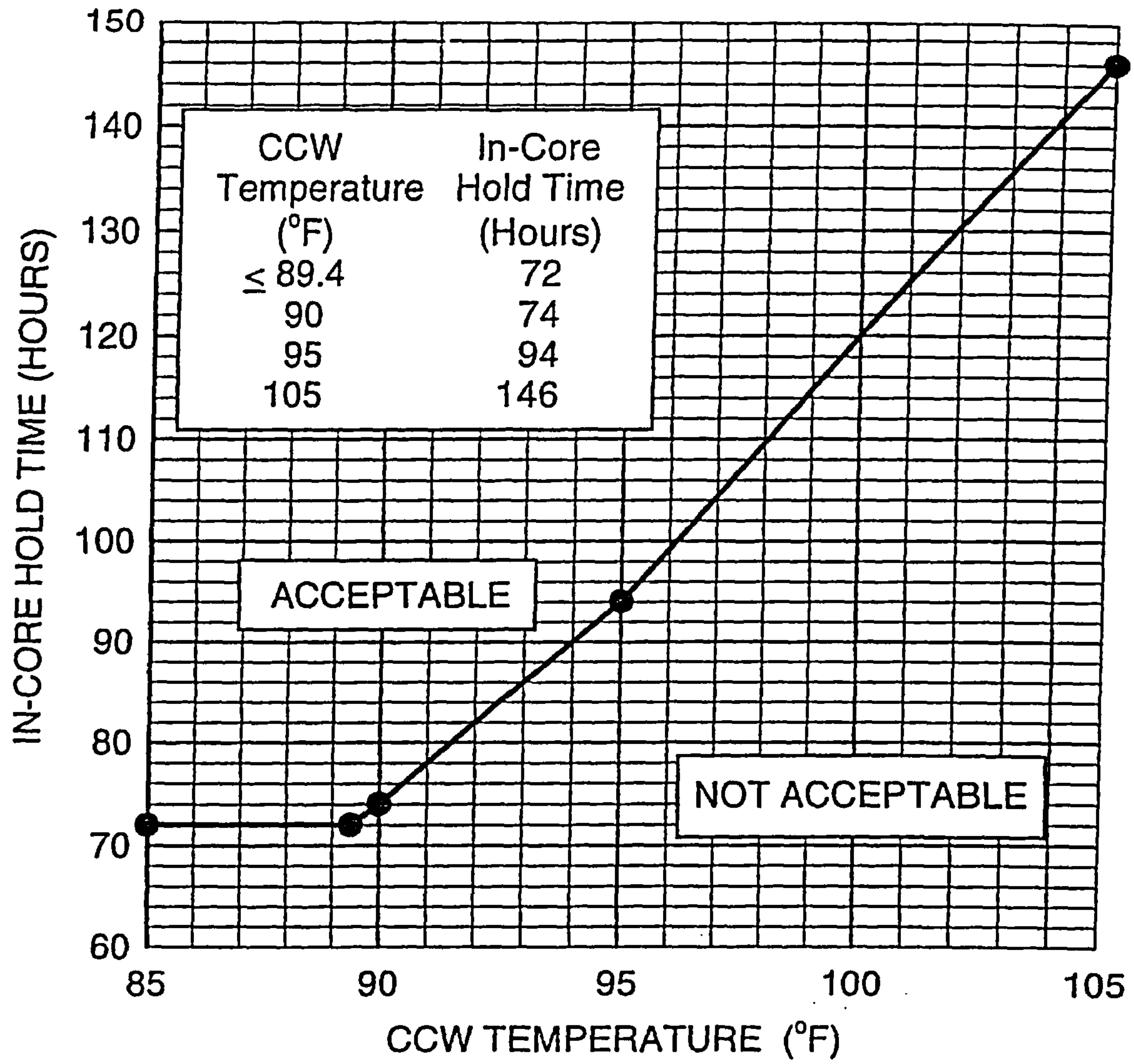
ACTION:

With the reactor subcritical for less than 72 hours, immediately suspend all movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for greater than 72 hours but not within the acceptable domain of Figure 3.9-1, immediately suspend movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3.1 The reactor shall be determined to have been subcritical for a period of time within the acceptable domain of Figure 3.9-1 by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.2 Prior to moving irradiated fuel from the reactor pressure vessel, and at least once every 12 hours during movement of irradiated fuel, verify the CCW temperature at the inlet to the Spent Fuel Pool Cooling System heat exchanger is within the acceptable domain of Figure 3.9-1.



Note: The use of linear interpolation between CCW temperatures reported above is acceptable to determine the minimum incore hold time.

FIGURE 3.9-1 REQUIRED IN-CORE HOLD TIME AS A FUNCTION OF COMPONENT COOLING WATER (CCW) TEMPERATURE

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 100 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 100 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 100 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

3/4.9.4 REACTOR BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The reactor building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the reactor building atmosphere to the outside atmosphere shall be either:

    1. Closed by an isolation valve, blind flange, or manual valve, or

    2. Be capable of being closed by an OPERABLE automatic Reactor Building Purge and Exhaust isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the reactor building.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the reactor building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required reactor building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic Reactor Building Purge and Exhaust isolation valve within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the reactor building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the Reactor Building Purge and Exhaust isolation valves per the applicable portions of Specification 4.6.4.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane and auxiliary hoist shall be used for movement of drive rods or fuel assemblies and shall be OPERABLE with:

a. The manipulator crane used for movement of fuel assemblies having:

   1. A minimum capacity of 3250 pounds, and

   2. An overload cut off limit less than or equal to 2700 pounds.

b. The auxiliary hoist used for latching and unlatching drive rods having:

   1. A minimum capacity of 3000 pounds, and

   2. A load indicator which shall be used to prevent lifting loads in excess of 1000 pounds.

APPLICABILITY: During movement of drive rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for crane and/or hoist OPERABILITY not satisfied, suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of drive rods and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6.1 Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 3250 pounds and demonstrating an automatic load cut off when the crane load exceeds 2700 pounds.

4.9.6.2 Each auxiliary hoist and associated load indicator used for movement of drive rods within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of such operations by performing a load test of at least 1250 pounds.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.1 At least one residual heat removal (RHR) loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no residual heat removal loop OPERABLE and in operation, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.1 At least one residual heat removal loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*The residual heat removal loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.7.2 Two independent Residual Heat Removal (RHR) loops shall be OPERABLE, and at least one RHR loop shall be in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required RHR loops OPERABLE, immediately initiate corrective action to return the required RHR loops to OPERABLE status or to establish greater than or equal to 23 feet of water above the reactor pressure flange, as soon as possible.

b. With no RHR loop in operation, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required RHR loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.7.2 At least one residual heat removal loop shall be verified in operation and circulating reactor coolant at a flow rate of greater than or equal to 2800 gpm at least once per 12 hours.

*Prior to initial criticality the residual heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.8 The Reactor Building Purge Supply and Exhaust Isolation System shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the Reactor Building Purge Supply and Exhaust Isolation System inoperable, close each of the Purge and Exhaust penetrations providing direct access from the reactor building atmosphere to the outside atmosphere. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 The Reactor Building Purge Supply and Exhaust Isolation System shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that Reactor Building Purge Supply and Exhaust isolation occurs on manual initiation, on a high radiation test signal from each of the containment radiation monitoring instrumentation channels, and by verifying that isolation occurs on the 36-inch lines of the Purge Supply and Exhaust Isolation System on a high radiation test signal from the reactor building manipulator crane area channels.

3/4.9.9 WATER LEVEL - REFUELING CAVITY AND FUEL TRANSFER CANAL

LIMITING CONDITION FOR OPERATION

3.9.9 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During movement of fuel assemblies or control rods within the reactor pressure vessel or the refueling cavity when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or control rods within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.9 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or control rods.

3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s),

APPLICABILITY: MODE 2.

ACTION:

a. With any full length control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to 30 gpm of a solution containing greater than or equal to 7000 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained less than or equal to 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.1 and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be less than or equal to 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 The Surveillance Requirements of the below listed Specifications (a. and b.) shall be performed at least once per 12 hours during PHYSICS TESTS:

a. Either Specifications 4.2.2.2 or 4.2.2.4 and Specification 4.2.2.5.

b. Specification 4.2.3.2.

3/4.10.3 PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specifications 3.1.1.3, 3.1.1.4, 3.1.3.1, 3.1.3.5 and 3.1.3.6 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at less than or equal to 25% of RATED THERMAL POWER, and

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

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

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

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.3.2 Each Intermediate and Power Range Channel shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

4.10.3.3 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 541°F at least once per 30 minutes during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1.1 may be suspended during the performance of start up and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed the P-7 Interlock Setpoint, and

b. The Reactor Trip Setpoints on the OPERABLE Intermediate and Power Range Channels are set less than or equal to 25% of RATED THERMAL POWER.

APPLICABILITY: During operation below the P-7 Interlock Setpoint.

ACTION:

With the THERMAL POWER greater than the P-7 Interlock Setpoint, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be less than P-7 Interlock Setpoint at least once per hour during start up and PHYSICS TESTS.

4.10.4.2 Each Intermediate, Power Range Channel and P-7 Interlock shall be subjected to an ANALOG CHANNEL OPERATIONAL TEST within 12 hours prior to initiating start up and PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.5 POSITION INDICATION SYSTEM - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.10.5 The limitations of Specification 3.1.3.3 may be suspended during the performance of individual full length (shutdown and control) rod drop time measurements provided;

a. Only one shutdown or control bank is withdrawn from the fully inserted position at a time, and

b. The rod position indicator is OPERABLE during the withdrawal of the rods.*

APPLICABILITY: MODES 3, 4 and 5 during performance of rod drop time measurements.

ACTION:

With the position indication system inoperable or with more than one bank of rods withdrawn, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.5 The above required rod position indication systems shall be determined to be OPERABLE within 24 hours prior to the start of and at least once per 24 hours thereafter during rod drop time measurements by verifying the demand position indication system and the rod position indication systems agree:

a. Within 12 steps when the rods are stationary, and

b. Within 24 steps during rod motion.

* This requirement is not applicable during the initial calibration of the rod position indication system provided (1) K_{eff} is maintained less than or equal to 0.95, and (2) only one shutdown or control rod bank is withdrawn from the fully inserted position at one time.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted by Amendment 104.
3.11.1.2 Deleted by Amendment 104.
3.11.1.3 Deleted by Amendment 104.

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Condensate Storage Tank
b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.1 Deleted by Amendment 104.
4.11.1.2 Deleted by Amendment 104.
4.11.1.3 Deleted by Amendment 104.

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

LIQUID HOLDUP TANKS

LIMITING CONDITION FOR OPERATION

3.11.1.1 Deleted by Amendment 104.
3.11.1.2 Deleted by Amendment 104.
3.11.1.3 Deleted by Amendment 104.

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to 10 curies, excluding tritium and dissolved or entrained noble gases.

a. Condensate Storage Tank
b. Outside Temporary Storage Tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.1 Deleted by Amendment 104.
4.11.1.2 Deleted by Amendment 104.
4.11.1.3 Deleted by Amendment 104.

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

SETTLING POND

LIMITING CONDITION FOR OPERATION

3.11.1.5 The quantity of radioactive material contained in each settling pond shall be limited by the following expression:

$$\frac{264}{V} \cdot \sum_{j} \frac{A_j}{C_j} < 1.0$$

excluding tritium and dissolved or entrained noble gases, where,

A_j = Pond inventory limit for single radionuclide "j", in curies.

C_j = 10 CFR 20, Appendix B, Table 2, column 2, concentration for single radionuclide "j", microcuries/ml.

V = design volume of liquid and slurry in the pond, in gallons.

264 = Conversion unit, microcuries/curie per milliliter/gallon.

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in the settling pond exceeding the above limit, immediately suspend all additions of radioactive material to the pond and within 48 hours reduce the pond contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.5 The quantity of radioactive material contained in each batch of slurry (used powdex resin) to be transferred to the settling ponds shall be determined to be within the above limit by analyzing a representative sample of the slurry, and batches to be transferred to the settling ponds shall be limited by the expression:

$$\sum_{j} \frac{Q_j}{C_j} < 1.0$$

SURVEILLANCE REQUIREMENTS (Continued)

where

Q_j = concentration of radioactive materials in wet, drained slurry (used powdex resin) for radionuclide "j" excluding tritium, dissolved or entrained noble gas and radionuclides with less than 8 day half-life, in microcuries per gram. The analysis shall include at least Ce-144, Cs-134, Cs-137, Sr-89, Sr-90, Co-58 and Co-60. Estimates of Sr-89, Sr-90, batch concentrations shall be based on the most recently available quarterly composite analyses.

C_j = 10 CFR 20, Appendix B, Table 2, column 2, concentration for single radionuclide "j", in microcuries/milliliter.

3/4.11.2 GASEOUS EFFLUENTS

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.1 Deleted by Amendment 104.
3.11.2.2 Deleted by Amendment 104.
3.11.2.3 Deleted by Amendment 104.
3.11.2.4 Deleted by Amendment 104.

3.11.2.5 The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of oxygen to within the limit within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than 4% by volume within 1 hour and less than or equal to 2% by volume within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.1 Deleted by Amendment 104.
4.11.2.2 Deleted by Amendment 104.
4.11.2.3 Deleted by Amendment 104.
4.11.2.4 Deleted by Amendment 104.

4.11.2.5 The concentration of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.9.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 131,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least one per 24 hours when radioactive materials are being added to the tank.

BASES FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The Bases contained in this section provide the bases for the specifications in Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specification of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.2 requires two independent ECCS subsystems to be OPERABLE and provides explicit ACTION requirements if one ECCS subsystem is inoperable. Under the requirements of Specification 3.0.3, if both of the required ECCS subsystems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Reactor Building Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the requirements of Specification 3.0.3, if both of the required Reactor Building Spray Systems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It 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. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is 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. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

4.0.3 Surveillance Requirement (SR) 4.0.3 establishes the flexibility to defer declaring affected equipment inoperable or an affected variable outside their specified limits when a surveillance has not been completed within the specified frequency. A delay period of up to 24 hours or up to the limit of the specified frequency, whichever is greater, applies from the point in time that it is discovered that the surveillance has not been performed in accordance with SR 4.0.2 and not at the time that the specified frequency was not met.

This delay period provides adequate time to complete surveillances that have been missed. This delay period permits the completion of a surveillance before complying with required Actions or other remedial measures that might preclude completion of the surveillance.

The basis for this delay period includes consideration of unit conditions, adequate planning, availability of personnel, the time required to perform the surveillance, the safety significance of the delay in completing the required surveillance, and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the requirements.

When a surveillance with a frequency based not on time intervals, but upon specified unit conditions, operating situations, or requirements of regulations (e.g., prior to entering MODE 1 after each fuel loading, or in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, etc.) is discovered to not have been performed when specified, SR 4.0.3 allows for the full delay period up to the specified frequency to perform the surveillance. However, since there is not a time interval specified, the missed surveillance should be performed at the first reasonable opportunity.

SR 4.0.3 provides a time limit for, and allowances for the performance of, surveillances that become applicable as a consequence of MODE changes imposed by required Actions.

Failure to comply with specified frequencies for SRs is expected to be an infrequent occurrence. Use of the delay period established by SR 4.0.3 is a flexibility, which is not intended to be used as an operational convenience to extend surveillance intervals.

While up to 24 hours or the limit of the specified frequency is provided to perform the missed surveillance, it is expected that the missed surveillance will be performed at the first reasonable opportunity. The determination of first reasonable opportunity should include consideration of the impact on plant risk (from delaying the surveillance as well as any plant configuration changes required or shutting the plant down to perform the surveillance) and impact on any analysis assumptions, in addition to unit conditions, planning, availability of personnel, and the time required to perform the surveillance. This risk impact should be managed through the program in place to implement 10 CFR 50.65(a)(4) and its implementing guidance, NRC Regulatory Guide 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants." This Regulatory Guide addresses consideration of temporary and aggregate risk impacts, determination of risk management action thresholds, and risk management action up to and including plant shutdown. The missed surveillance should be treated as an emergent condition as discussed in the Regulatory Guide. The risk evaluation may use quantitative, qualitative, or blended methods. The degree of depth and rigor of the evaluation should be commensurate with the importance of the component. Missed surveillances for important components should be analyzed quantitatively. If the results of the risk evaluation determine the risk increase is significant, this evaluation should be used to determine the safest course of action. All missed surveillances will be placed in the Corrective Action Program.

If a surveillance is not completed within the allowed delay period, then the equipment is considered inoperable or the variable is considered outside the specified limits and the Allowed Outage Time (AOT) for the required Action for the applicable LCO conditions begin immediately upon expiration of the delay period. If a surveillance is failed within the delay period, then the equipment is inoperable, or the variable is considered outside the specified limits and the AOT of the required Action for the applicable LCO begin immediately upon the failure of the surveillance.

Completion of the surveillance within the delay period allowed by this specification, or within the AOT of the Action, restores compliance with SR 4.0.1.

4.0.4 This specification 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 provision 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.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

APPLICABILITY

BASES

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. In MODES 1 and 2 the most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 1.77 percent delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. In MODES 3, 4 and 5 the most limiting accident is a boron dilution accident. The SHUTDOWN MARGIN is varied as a function of average RCS boron concentration in order to provide adequate protection in these MODES.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC value equivalent to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections

MODERATOR TEMPERATURE COEFFICIENT (Continued)

involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperature at RATED THERMAL POWER conditions. This value of the MDC was then transformed into the limiting End of Cycle Life (EOL) MTC value. The 300 ppm surveillance limit MTC value 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 EOL MTC value.

The surveillance requirements for measurement of the MTC at the beginning and near the end of the fuel cycle are adequate to confirm that the MTC remains within its limits since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 551°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid transfer pumps, and 5) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two boron injection flow paths are required to ensure single functional capability in the event an assumed failure renders one of the flow paths inoperable. The boration capability of either flow path is sufficient to provide the required SHUTDOWN

BORATION SYSTEMS (Continued)

MARGIN from expected operating conditions of 1.77% delta k/k or as required by Figure 3.1-3 after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs from full power equilibrium xenon conditions and is satisfied by 13269 gallons of 7000 ppm borated water from the boric acid storage tanks or 98631 gallons of 2300 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide the required SHUTDOWN MARGIN of 1 percent delta k/k or as required by Figure 3.1-3 after xenon decay and cooldown from 200°F to 140°F. This condition is satisfied by either 2000 gallons of 7000 ppm borated water from the boric acid storage tanks or 23266 gallons of 2300 ppm borated water from the refueling water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of rod misalignment on associated accident analyses. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

MOVABLE CONTROL ASSEMBLIES (Continued)

For purposes of determining compliance with Technical Specification 3.1.3.1, any inoperability of full length control rod(s), due to being immovable, invokes ACTION statement "a".

The intent of Technical Specification 3.1.3.1 ACTION statement "a" is to ensure that before leaving ACTION statement "a" and utilizing ACTION statement "c" that the rod urgent failure alarm is illuminated or that an obvious electrical problem is detected in the rod control system by minimal electrical troubleshooting techniques. Expeditious action will be taken to determine if rod immovability is due to an electrical problem in the rod control system.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. In addition, those safety analyses affected by a misaligned rod are reevaluated to confirm that the results remain valid during future operation.

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 551°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

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 calculated DNBR in the core at or above the design limit 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.

3/4.2.1 AXIAL FLUX DIFFERENCE

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(z)$ upper bound envelope of the F_Q limit specified in the CORE OPERATING LIMITS REPORT (COLR) times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

The limits on AFD will be provided in the COLR per Technical Specification 6.9.1.11.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

AXIAL FLUX DIFFERENCE (Continued)

At power levels below APL^{ND}, the limits on AFD are defined in the COLR consistent with the Relaxed Axial Offset Control (RAOC) operating procedure and limits. These limits were calculated in a manner such that expected operational transients, e.g., load follow operations, would not result in the AFD deviating outside of those limits. However, in the event such a deviation occurs, the short period of time allowed outside of the limits at reduced power levels will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prevent operation in the vicinity of the APL^{ND} power level.

At power levels greater than APL^{ND}, two modes of operation are permissible; (1) RAOC, the AFD limit of which are defined in the COLR and (2) Base Load operation, which is defined as the maintenance of the AFD within COLR specifications band about a target value. The RAOC operating procedure above APL^{ND} is the same as that defined for operation below APL^{ND}. However, it is possible when following extended load following maneuvers that the AFD limits may result in restrictions in the maximum allowed power or AFD in order to guarantee operation with $F_Q(z)$ less than its limiting value. To allow operation at the maximum permissible power level the Base Load operating procedure restricts the indicated AFD to relatively small target band (as specified in the COLR) and power swings ($APL^{ND} \leq$ power $\leq APL^{BL}$ or 100% Rated Thermal Power, whichever is lower). For Base Load operation, it is expected that the plant will operate within the target band. Operation outside of the target band for the short time period allowed will not result in significant xenon redistribution such that the envelope of peaking factors would change sufficiently to prohibit continued operation in the power region defined above. To assure there is no residual xenon redistribution impact from past operation on the Base Load operation, a 24-hour waiting period at a power level above APL^{ND} and allowed by RAOC is necessary. During this time period load changes and rod motion are restricted to that allowed by the Base Load procedure. After the waiting period extended Base Load operation is permissible.

The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are: (1) outside the allowed delta-I power operating space (for RAOC operation), or (2) outside the allowed delta-I target band (for Base Load operation). These alarms are active when power is greater than: (1) 50% of RATED THERMAL POWER (for RAOC operation), or (2) APL^{ND} (for Base Load operation). Penalty deviation minutes for Base Load operation are not accumulated based on the short period of time during which operation outside of the target band is allowed.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor, RCS flowrate, and nuclear enthalpy rise hot channel factor ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 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.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

$F^{N}_{\Delta H}$ will be maintained within its limits provided conditions a. through d. above are maintained. As noted on the RCS Total Flow Rate Versus R figure in the CORE OPERATING LIMITS REPORT (COLR), RCS flow rate and power may be "traded off" against one another (i.e., a low measured RCS flow rate is acceptable if core power is also low) to ensure that the calculated DNBR will not be below the design DNBR value. The relaxation of $F^{N}_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits.

R, as calculated in 3.2.3 and used in the RCS Total Flow Rate Versus R figure in the COLR, accounts for $F^{N}_{\Delta H}$ less than or equal to the $F^{RTP}_{\Delta H}$ limit specified in the COLR. This value is used in the various accident analyses where $F^{N}_{\Delta H}$ influences parameters other than DNBR, e.g., peak clad temperature and thus is the maximum "as measured" value allowed.

Margin is maintained between the safety analysis limit DNBR and the design limit DNBR. This margin is more than sufficient to offset any rod bow penalty and transition core penalty. The remaining margin is available for plant design flexibility.

When a 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.

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

For measurements obtained using the Power Distribution Monitoring System (PDMS), the appropriate measurement uncertainty is determined using the measurement uncertainty methodology contained in WCAP-12472-P-A. The cycle and plant specific uncertainty calculation information needed to support the PDMS calculation is contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty, and apply a 3% allowance for manufacturing tolerance.

The hot channel factor $F_Q^M(z)$ is measured periodically and increased by a cycle and height dependent power factor appropriate to either RAOC or Base Load operation, W(z) or $W(z)_{BL}$, to provide assurance that the limit on the hot channel factor, $F_Q(z)$ is met. W(z) accounts for the effects of normal operation transients and was determined from expected power control maneuvers over the full range of burnup conditions in the core. $W(z)_{BL}$ accounts for the more restrictive operating limits allowed by Base Load operation which result in less severe transient values. If two most recent $F_Q(z)$ evaluations show an increase in the maximum value of $\left[\frac{F_Q^M(z)}{K(z)}\right]$ over the core height (z), it is not guaranteed that $F_Q^M(z)$ will remain within the transient limit during the following surveillance interval. Technical Specification Surveillance Requirement 4.2.2 requires that $F_Q^M(z)$ be increased by a penalty factor as specified in the COLR and compared to the transient $F_Q(z)$ limit. If there is insufficient margin, i.e., this value exceeds the limit, the $F_Q^M(z)$ must be measured once per 7 EFPD until either $F_Q^M(z)$ increased by the penalty factor is within the transient limit, or two successive power distribution measurements indicate the maximum value of $\left[\frac{F_Q^M(z)}{K(z)}\right]$ over the core height (z) has not increased. The W(z) and $W(z)_{BL}$ functions described above for normal operation are specified in the CORE OPERATING LIMITS REPORT (COLR) per Specification 6.9.1.11.

When RCS flow rate is measured, no additional allowances are necessary prior to comparison with the limits of the RCS Total Rate Versus R figure in the COLR. Measurement errors of 2.1% for RCS total flow rate, including 0.1% for feedwater venturi fouling, have been allowed for in determining the limits of RCS Total Flow Rate Versus R Figure in the COLR.

For $F_{\Delta H}^N$ measurements obtained from a full core flux map taken with the incore detector flux mapping system, a 4% measurement uncertainty allowance should be applied to the measured $F_{\Delta H}^N$ value prior to comparison with the limits of the RCS Total Flow Rate Versus R Figure in the COLR. The appropriate measurement uncertainty for $F_{\Delta H}^N$ measurements obtained using the Power Distribution Monitoring System (PDMS) is determined using the uncertainty methodology described in WCAP-12472-P-A. The cycle and plant specific uncertainty calculation information needed to support the PDMS uncertainty calculation is

HEAT FLUX HOT CHANNEL FACTOR and RCS FLOWRATE and NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR (Continued)

contained in the COLR. The PDMS will automatically calculate and apply the correct measurement uncertainty to the measured $F^N_{\Delta H}$ value.

The 12-hour periodic surveillance of indicated RCS flow is sufficient to detect only flow degradation which could lead to operation outside the acceptable region of operation specified on the RCS Total Flow Rate Versus R F.igure in the COLR.

3/4.2.4 QUADRANT POWER TILT RATIO

The quadrant power tilt power ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainty in F_Q is depleted. The limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The two hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action does not correct the tilt, the margin for uncertainty on F_Q is reinstated by reducing the maximum allowed power by 3 percent for each percent of tilt in excess of 1.0.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or a core power distribution measurement are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of 4 symmetric thimbles. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum of DNBR in the core at or above the design limit throughout each analyzed transient. The maximum indicated T_{avg} limit of 589.2°F and the minimum indicated pressure limit of 2206 psig correspond to analytical limits of 591.4°F and 2185 psig respectively, read from control board indications.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

3/4.3.1 and 3/4.3.2 REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor Protection System and Engineered Safety Feature Actuation System Instrumentation and interlocks ensure that 1) the associated action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoints, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functions capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses. The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability. Specified surveillance and surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for Reactor Protection Instrumentation System," and supplements to that report. Surveillance intervals and out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The Engineered Safety Feature Actuation System Instrumentation Trip Setpoints specified in Table 3.3-4 are the nominal values at which the bistables are set for each functional unit. A setpoint is considered to be adjusted consistent with the nominal value when the "as measured" setpoint is within the band allowed for calibration accuracy.

To accommodate the instrument drift assumed to occur between operational tests and the accuracy to which setpoints can be measured and calibrated, Allowable Values for the setpoints have been specified in Table 3.3-4. Operation with setpoints less conservative than the Trip Setpoint but within the Allowable Value is acceptable since an allowance has been made in the safety analysis to accommodate this error.

The methodology to derive the trip setpoints is based upon combining all of the uncertainties in the channels. Inherent to the determination of the trip setpoints are the magnitudes of these channel uncertainties. Sensor and rack instrumentation utilized in these channels are expected to be capable of operating within the allowances of these uncertainty magnitudes. Rack drift in excess of the Allowable Value exhibits the behavior that the rack has not met its allowance. Being that there is a small statistical chance that this

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

will happen, an infrequent excessive drift is expected. Rack or sensor drift, in excess of the allowance that is more than occasional, may be indicative of more serious problems and should warrant further investigation.

The measurement of response time at the specified frequencies provides assurance that the reactor trip and the engineered safety feature actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable. Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Response time may be verified by actual response time tests in any series of sequential, overlapping, or total channel measurements, or by the summation of allocated sensor, signal processing, and actuation logic response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise or power interrupt tests), (2) in place, onsite, or offsite (e.g., vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements," provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types must be demonstrated by test.

WCAP-14036-P-A, Revision 1, "Elimination of Periodic Response Time Tests," provides the basis and methodology for using allocated signal processing and actuation logic response times in the overall verification of the protection system channel response time. The allocations for sensor, signal conditioning, and actuation logic response times must be verified prior to placing the component into operational service and re-verified following maintenance or modification that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for the repair are of the same type and value. Specific components identified in the WCAP may be replaced without verification testing. One example where response time could be affected is replacing the sensing element of a transmitter.

Westinghouse letter CGE-00-018, dated March 28, 2000, provided an evaluation of the Group 05 (11NLP and 6NSA) 7300 process cards. These cards were revised after the submittal of WCAP-14036, Revision 1. This letter concluded that the failure modes and effects analysis (FMEA) performed for the older versions of these cards and documented in WCAP-14036-P-A, Revision 1, is applicable for these Group 05 cards. The bounding time response values determined by test and evaluation and reported in the WCAP are valid for these redesigned cards.

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

The Engineered Safety Features response times specified in Table 3.3-5 which include sequential operation of the RWST and VCT valves (Notes 2 and 3) are based on values assumed in the non-LOCA safety analyses. These analyses are for injection of borated water from the RWST. Injection of borated water is assumed not to occur until the VCT charging pump suction isolation valves are closed following opening of the RWST charging pumps suction valves. When the sequential operation of the RWST and VCT valves is not included in the response times (Note 1) the values specified are based on the LOCA analyses. The LOCA analyses take credit for injection flow regardless of the source. Verification of the response times specified in Table 3.3-5 will assure that the assumptions used for the LOCA and non-LOCA analyses with respect to the operation of the VCT and RWST valves are valid.

The Engineered Safety Features Actuation System senses selected plant parameters and determines whether or not predetermined limits are being exceeded. If they are, the signals are combined into logic matrices sensitive to combinations indicative of various accidents, events, and transients. Once the required logic combination is completed, the system sends actuation signals to those engineered safety features components whose aggregate function best serves the requirements of the condition. As an example, the following actions may be initiated by the Engineered Safety Features Actuation System to mitigate the consequences of a steam line break or loss of coolant accident 1) safety injection pumps start and automatic valves position, 2) reactor trip, 3) feedwater isolation, 4) startup of the emergency diesel generators, 5) containment spray pumps start and automatic valves position, 6) containment isolation, 7) steam line isolation, 8) turbine trip, 9) auxiliary feedwater pumps start and automatic valves position, 10) containment cooling fans start and automatic valves position, 11) essential service water pumps start and automatic valves position, and 12) control room isolation and ventilation systems start.

Several automatic logic functions included in this specification are not necessary for Engineered Safety Feature System actuation but their functional capability at the specified setpoints enhances the overall reliability of the Engineered Safety Features functions. These automatic actuation Systems are purge and exhaust isolation from high containment radioactivity, turbine trip and feedwater isolation from steam generator high-high water level, initiation of emergency feedwater on a trip of the main feedwater pumps, automatic transfer of the suctions of the emergency feedwater pumps to service water on low suction pressure, and automatic opening of the containment recirculation sump suction valves for the RHR and spray pumps on low-low refueling water storage tank level.

REACTOR TRIP AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (continued)

The service water response time includes: 1) the start of the service water pumps and, 2) the service water pumps discharge valves (3116A,B,C-SW) stroking to the fully opened position. This condition of the valves assures that flow will become established through the component cooling water heat exchanger, diesel generator coolers, HVAC chiller, and to the suction of the service water booster pumps when these components are placed in-service. Prior to this time, the flow is rapidly approaching required flow and sufficient pressure is developed as valves finish their stroke. Each of the above-listed components will be starting to perform their accident mitigation function, either directly or indirectly depending upon the use of the component, and will be operational within the service water response time of 71.5/81.5 seconds[1/]. Only the service water booster pumps have a direct impact on the accident analysis via the RBCUs' heat removal capability as discussed below.

The RBCU response time includes: 1) the start of the RBCU fan and the service water booster pumps and, 2) all the service water valves which must be driven to the fully opened or fully closed position. This condition of the valves allows the flow to become fully established through the RBCU. Prior to this time, the flow is rapidly approaching required flow as the valves finish their stroke. Although the RBCU would be removing heat throughout the Engineered Safety Features response time, the accident analysis does not assume heat removal capability from 0 to 71.5 seconds [2/] because the industrial cooling water system is not completely isolated until 71.5 seconds. A linear ramp increase from 95% full heat removal capability to 100% full heat removal capability is assumed by the accident analysis to start at 71.5 seconds and end at 86.5 seconds[3/]. Full heat removal capability is assumed at 86.5 seconds based on the position of the valve 3107-SW.

1/ Total time is 1.5 second instrument response after setpoint is reached, plus 10 seconds diesel generator start, plus 10 seconds to reach service water pump start and begin 3116-SW opening via Engineered Safety Features Loading Sequencer, plus 60 seconds stroke time for 3116-SW. During this total time, the service water pumps start and the service water pump discharge valve begins to open at 11.5 seconds and the pump discharge valve is fully open at 71.5 seconds without a diesel generator start required and 21.5 seconds and 81.5 seconds including a diesel generator start.

2/ Total time is 1.5 second instrument response after setpoint is reached, plus 10 second diesel start plus 60 seconds* for valves to isolate industrial cooling water system.

3/ Total time is 1.5 second instrument response after setpoint is reached, plus 10 second diesel generator start plus 75 seconds to stroke valves 3107A, B-SW.

* During this time period, the Engineered Safety Features Loading Sequencer starts the RBCU fans at 25 seconds and service water booster pumps at 30 seconds after the valves begin to stroke.

REACTOR PROTECTION SYSTEM AND ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION (Continued)

The Engineered Safety Feature Actuation System interlocks perform the following functions:

P-4 Reactor tripped - Actuates turbine trip, closes main feedwater valves on T_{avg} below setpoint, prevents the opening of the main feedwater valves which were closed by a safety injection or high steam generator water level signal, allows safety injection block so that components can be reset or tripped.

Reactor not tripped - prevents manual block of safety injection.

P-11 On increasing pressurizer pressure, P-11 automatically reinstates safety injection actuation on low pressurizer pressure. On decreasing pressure, P-11 allows the manual block of safety injection actuation on low pressurizer pressure.

P-12 On increasing primary coolant loop temperature, P-12 automatically reinstates safety injection actuation and steam line isolation on low steam line pressure, and removes a blocking signal from the steam dump system. On decreasing primary coolant loop temperature, P-12 allows the manual block of safety injection actuation and steam line isolation on low steam line pressure and automatically provides a blocking signal to the steam dump system.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 MOVABLE INCORE DETECTORS

The OPERABILITY of the movable incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The OPERABILITY of this system is demonstrated by irradiating each detector used and determining the acceptability of its voltage curve.

For the purpose of measuring $F_Q(Z)$ or $F^N_{\Delta H}$ a full incore flux map is used. Quarter-core flux maps, as defined in WCAP-8648, June 1976, may be used in recalibration of the excore neutron flux detection system, and full incore flux maps or symmetric incore thimbles may be used for monitoring the QUADRANT POWER TILT RATIO when one Power Range Channel is inoperable.

3/4.3.3.3 SEISMIC INSTRUMENTATION

Deleted.

3/4.3.3.4 METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23, "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The PAM Instrumentation LCO provides OPERABILITY requirements for Regulatory Guide 1.97 Type A monitors, which provide information required to perform certain manual actions specified in the Emergency Operating Procedures. These manual actions ensure that a system can accomplish its safety function and are credited in the safety analyses. Additionally, this LCO addresses Regulatory Guide 1.97 instruments that have been designated Category I, non-Type A.

The OPERABILITY of the PAM instrumentation ensures there is sufficient information available on selected unit parameters to monitor and assess unit status following an accident.

LCO 3.3.3.6 requires two OPERABLE channels for most Functions. Two OPERABLE channels ensure no single failure prevents operators from getting the information necessary for them to determine the safety status of the unit, and to bring the unit to and maintain it in a safe condition following an accident.

Furthermore, OPERABILITY of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

Type A and Category I variables are required to meet Regulatory Guide 1.97 Category I design and qualification requirements for seismic and environmental qualification, single failure criterion, utilization of emergency standby power, immediately accessible display, continuous readout, and recording of display.

The following are discussions of specified instrument functions listed in Table 3.3-10.

1. & 2. Reactor Building Pressure

Reactor Building Pressure is provided for verification of RCS and containment OPERABILITY. Reactor Building Pressure is used to verify closure of main steam isolation valves (MSIVs), and containment spray Phase B isolation. Other manual actions based on Reactor Building Pressure include: stopping the RCPs, stopping containment spray pumps, and starting RHR pumps. Reactor Building Pressure indications are also required to calculate reactor vessel vent times.

3. Reactor Building Radiation Level

Reactor Building Radiation Level is provided to monitor the potential of significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. Reactor Building Radiation Level is used to determine if a high energy line break (HELB) has occurred and whether the event is inside or outside of containment.

4. Reactor Building Hydrogen Concentration

Reactor Building Hydrogen Concentration Monitors are provided to detect high hydrogen concentration conditions that represent a potential for containment breach from a hydrogen explosion. This variable is also important in verifying the adequacy of mitigating actions. Reactor Building Hydrogen concentration is used by the operator to calculate reactor vessel vent time and is monitored as a criterion for continuing vessel venting when attempting to collapse voids.

5. Reactor Building/RHR Sump Water Level (Wide Range)

Reactor Building/RHR Sump Water Level is provided for verification and long term surveillance of RCS integrity. Reactor Building/RHR Sump Water Level is used to determine: containment sump level accident diagnosis; when to begin the recirculation procedure; and whether to terminate SI, if still in progress.

6. & 7. Reactor Coolant System (RCS) Hot and Cold Leg Temperatures

RCS Hot and Cold Leg Temperatures variables provide verification of core cooling and long term RCS surveillance. In addition, RCS cold leg temperature is used in conjunction with RCS hot leg temperature to verify the unit conditions necessary to establish natural circulation in the RCS.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

8. Reactor Coolant Pressure

RCS pressure provides verification of core cooling and RCS integrity long term surveillance. RCS pressure is used to verify delivery of SI flow to RCS from at least one train when the RCS pressure is below the pump shutoff head. RCS pressure is also used to verify closure of manually closed spray line valves and pressurizer power operated relief valves (PORVs). RCS pressure can be used: to determine whether to terminate actuated SI or to reinitiate stopped SI; to determine when to reset SI and shut off low head SI; to manually restart low head SI; as reactor coolant pump (RCP) trip criteria; and to make a determination on the nature of the accident in progress and where to go next in the procedure. RCS pressure is also related to three decisions about depressurization. They are: to determine whether to proceed with primary system depressurization; to verify termination of depressurization; and to determine whether to close accumulator isolation valves during a controlled cooldown/depressurization. A final use of RCS pressure is to determine whether to operate the pressurizer heaters. RCS pressure is used by the operator to monitor the cooldown of the RCS following a steam generator tube rupture (SGTR) or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting steam generator (SG) pressure or level, would use this indication. Furthermore, RCS pressure is one factor that may be used in decisions to terminate RCP operation.

9. Pressurizer Level

Pressurizer Level is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition.

10. Reactor Coolant System Subcooling Margin

RCS hot and cold leg temperatures are used to determine RCS subcooling margin. RCS subcooling margin will allow termination of safety injection (SI), if still in progress, or reinitiation of SI if it has been stopped. RCS subcooling margin is also used for unit stabilization and cooldown control.

11. Reactor Vessel Level

Reactor Vessel Level is provided for verification and long term surveillance of core cooling. It is also used for accident diagnosis and to determine reactor coolant inventory adequacy.

Reactor Vessel Level Monitoring provides direct measurement of collapsed liquid level above the fuel alignment plate. Collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of collapsed water level is selected because it is a direct indication of the water inventory.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

12. Core Exit Temperature

Core Exit Temperature is provided for verification and long term surveillance of core cooling. Core Exit Temperature is used to determine whether to terminate SI, if still in progress, or to reinitiate SI if it has been stopped. Core Exit Temperature is also used for unit stabilization and cooldown control.

Two OPERABLE channels of Core Exit Temperature, in each quadrant, provide indication of radial distribution of coolant temperature rise across representative regions of the core. Power distribution symmetry was considered in determining the specific number and locations provided for diagnosis of local core problems. Two randomly selected thermocouples are not sufficient to meet the two thermocouples per channel requirement in any quadrant. The two thermocouples in each channel must meet the additional requirement that one is located near the center of the core and the other near the core perimeter, such that the pair of Core Exit Temperatures indicate the radial temperature gradient across their core quadrant. Two sets of two thermocouples ensure a single failure will not disable the ability to determine the radial temperature gradient.

13. Neutron Flux

Neutron Flux indication is provided to verify reactor shutdown. Neutron flux is used for accident diagnosis, verification of subcriticality, and diagnosis of positive reactivity insertion.

14. Steam Line Pressure

Steam Line Pressure is used to: identify and isolate a faulted steam generator; maintain an adequate reactor heat sink; verify that EFW to steam generator associated with pipe rupture is isolated; and monitor secondary side steam pressure to: verify operation of pressure control steam dump system, monitor RCS cooldown rate, and maintain plant in cold shutdown condition.

ACCIDENT MONITORING INSTRUMENTATION (Continued)

15. & 16. Steam Generator Water Level

SG Water Level is provided to monitor operation of decay heat removal via the SGs. Temperature compensation of this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is input to the unit computer, a control room indicator, and the Emergency Feedwater Control System.

SG Water Level is used to: identify the faulted SG following a tube rupture; verify that the intact SGs are an adequate heat sink for the reactor; determine the nature of the accident in progress (e.g., verify an SGTR); and verify unit conditions for termination of SI during secondary unit HELBs outside containment. Operator action is based on control room indication of SG level. The RCS response during a design basis small break LOCA depends on the break size. For a certain range of break sizes, SG-condenser mode of heat transfer is necessary to remove decay heat. Operator action is required to manually raise and control SG level to establish heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated level reaches its setpoint.

17. Emergency Feedwater Flow

EFW Flow is provided to monitor operation of decay heat removal via the SGs. EFW Flow to each SG is determined from a differential pressure measurement calibrated for a range of 0 gpm to 1000 gpm. Redundant monitoring capability is provided by two independent trains of instrumentation for each SG. Each differential pressure transmitter provides an input to a control room indicator and the unit computer. Since the primary indication used by the operator during an accident is the control room indicator, the PAM specification deals specifically with this portion of the instrument channel. EFW flow is used three ways: to verify delivery of EFW flow to the SGs; to determine whether to terminate SI if still in progress, in conjunction with SG water level (narrow range); and to regulate EFW flow so that the SG tubes remain covered. Operator action is required to throttle EFW flow during an SLB accident to prevent the EFW pumps from operating in runout conditions. EFW flow is also used by the operator to verify that the EFW System is delivering the correct flow to each SG. However, the Primary indication used by the operator to ensure an adequate inventory is SG level.

18. Refueling Water Storage Tank (RWST) Level

RWST Level is provided to ensure water supply for ECCS. RWST low level indications are used by the operator as the basis for aligning ECCS suction to the containment sump and stopping all pumps taking suction from the RWST on low-low level.

3/4.3.3.9 EXPLOSIVE GAS MONITORING INSTRUMENTATION

This instrumentation includes provisions for monitoring and controlling the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.11 POWER DISTRIBUTION MONITORING SYSTEM (PDMS)

The Power Distribution Monitoring System (PDMS) provides core monitoring of the limiting parameters. The PDMS continuous core power distribution measurement methodology begins with the periodic generation of a highly accurate 3-D nodal simulation of the current reactor power distribution. The simulated reactor power distribution is then continuously adjusted by nodal and thermocouple calibration factors derived from an incore power distribution measurement obtained using the incore movable detectors to produce a highly accurate power distribution measurement. The nodal calibration factors are updated at least once every 180 Effective Full Power Days (EFPD). Between calibrations, the fidelity of the measured power distribution is maintained via adjustment to the calibrated power distribution provided by continuously input plant and core condition information. The plant and core condition data utilized by the PDMS is cross checked using redundant information to provide a robust basis for continued operation. The loop inlet temperature is generated by averaging the respective temperatures from each of the loops, excluding any bad data. The core exit thermocouples provide many readings across the core and by the nature of their usage with the PDMS, smoothing of the measured data and elimination of bad data is performed with the Surface Spline fit. PDMS uses the NIS Power Range excore detectors to provide information on the axial power distribution. Hence, the PDMS averages the data from the four Power Range excore detectors and eliminates any bad excore detector data.

The bases for the operability requirements of the PDMS is to provide assurance of the accuracy and reliability of the core parameters measured and calculated by the PDMS core power distribution monitor function. These requirements fall under four categories:

1. Assure an adequate number of operable critical sensors.
2. Assure sufficiently accurate calibration of these sensors.
3. Assure an adequate calibration data base regarding the number of data sets.
4. Assure the overall accuracy of the calibration.

POWER DISTRIBUTION MONITORING SYSTEM (PDMS) (Continued)

The minimum number of required plant and core condition inputs includes the following:

1. Control Bank Positions.
2. At least 50% of the cold leg temperatures.
3. At least 75% of the signals from the Power Range excore detector channels (comprised of a top and bottom detector section).
4. Reactor Power Level.
5. A minimum number and distribution of operable core exit thermocouples.
6. A minimum number and distribution of measured fuel assembly power distribution information obtained using the incore movable detectors is incorporated in the nodal model calibration information.

The sensor calibration of items 1., 2., 3., and 4. above are covered under other specifications. Calibration of the core exit thermocouples is accomplished in two parts. The first being a sensor specific correction to K-type thermocouple temperature indications based on data from a cross calibration of the thermocouple temperature indications to the average RCS temperature measured via the RTDs under isothermal RCS conditions. The second part of the thermocouple calibration is the generation of thermocouple flow mixing factors which cause the radial power distribution measured via the thermocouples to agree with the radial power distribution from a full core flux map measured using the incore movable detectors. This calibration is updated at least once every 180 EFPD.

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 in the core at or above the 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 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either RHR or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single RHR loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two RHR loops be OPERABLE.

The operation of one Reactor Coolant Pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump with one or more RCS cold legs less than or equal to 300°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either (1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or (2) by restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 50°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 420,000 lbs per hour of saturated steam at the valve set point plus 3% accumulation. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating RHR loop, connected to the RCS, provides overpressure relief capability and will prevent RCS over-pressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss of load assuming no reactor trip until the first Reactor Protective System trip set point is reached (i.e., no credit is taken for a direct reactor trip on the loss of load) and also assuming no operation of the power operating relief valves or steam dump valves.

Demonstration of the safety valves' lift settings will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady state envelope of operation assumed in the SAR. The limit is consistent with the initial SAR assumptions. The 12 hour periodic surveillance is sufficient to ensure that the parameter is restored to within its limit following expected transient operation. The maximum water volume also ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

3/4.4.4 RELIEF VALVES (PORVs)

The pressurizer 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. The PORVs and block valves may be used to depressurize the RCS when normal pressurizer spray is unavailable. Operation of the air operated PORVs minimizes the undesirable opening of the spring loaded pressurizer code safety valves. Each PORV has a remotely controlled motor-operated block valve to provide a positive shutoff capability should a relief valve become inoperable. The series arrangement of the PORV and its associated block valve permit surveillance while at power.

RELIEF VALVES (PORVs) (Continued)

The PORVs, their block valves, and their controls are powered from buses that normally receive offsite power but are also capable of being powered from emergency power sources. Two PORVs and their associated block valves are powered from two separate safety trains. By maintaining two PORVs and their associated block valves OPERABLE, redundant capability to perform their design function is maintained.

Plant operators employ the PORVs to depressurize the RCS in response to certain plant transients if normal pressurizer spray is not available. For the Steam Generator Tube Rupture (SGTR) event, the safety analysis assumes that manual operator actions are required to mitigate the event. A loss of offsite power is assumed to accompany the event, and thus, normal pressurizer spray is unavailable to reduce RCS pressure. The PORVs are assumed to be used for RCS depressurization, which is one of the steps performed to equalize the primary and secondary pressures in order to terminate the primary to secondary break flow and the radioactive releases from the affected steam generator.

Credit is taken for the PORVs in safety analyses of events that result in increasing RCS pressure where departure from nucleate boiling ratio (DNBR) criteria are critical. By assuming PORV manual actuation, the primary pressure remains below the high pressurizer pressure trip setpoint, thus the DNBR calculation is more conservative. Events that assume this condition include a turbine trip and the loss of normal feedwater.

One PORV that is capable of manual operation has sufficient capacity to perform its function to depressurize the RCS and mitigate the effects of a postulated event. Two PORVs that are OPERABLE or capable of manual operation provide adequate redundancy.

Operating the PORV and block valve through one complete cycle verifies that the valve and its associated supporting systems are capable of manual operation.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 gallons per day per steam generator). Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 150 gallons per day per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage-type degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to 10CFR50.72(b)2(i) 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.

Correction letter of 12-15-2003

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The CONTROLLED LEAKAGE limitation restricts operation when the total flow supplied to the reactor coolant pump seals exceeds 33 GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of 2235 psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.

10CFR50.2, 10CFR50.55a(c), and GDC 55 of 10CFR50, Appendix A define RCS PIVs as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB) which separate the high pressure RCS from an attached low pressure system. During their service lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV leakage LCO allows leakage through these valves in amounts that do not compromise safety.

The PIV LEAKAGE limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of IDENTIFIED LEAKAGE governed by LCO 3.4.6.2, "REACTOR COOLANT SYSTEM, OPERATIONAL LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by water inventory balance (SR 4.4.6.2.1.d). A known component of the identified leakage before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing. Leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other PIV is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low-pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting system are degraded or degrading. Excessive PIV leakage could lead to overpressure of the low-pressure piping or components, potentially resulting in a loss of coolant accident (LOCA) outside of containment.

OPERATIONAL LEAKAGE (Continued)

The PIV leakage limit is 0.5 GPM per nominal inch of valve size with a maximum limit of 5 GPM. The NRC, through NUREG-1431, has endorsed this PIV leakage rate limit.

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

Leakage from the RCS Pressure Isolation Valves may be identified by surveillance testing performed during plant heatup or cooldown above 2000 psig and may be adjusted to obtain the leakage value at 2235 ± 20 psig using calculation guidance provided by ASME Code, Section XI, Part OM-10.

The maximum allowed steam generator tube leakage of 450 GPD (3 steam generators with 150 GPD each) for all steam generators not isolated from the RCS ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 150 GPD per steam generator limit preserves the assumptions used in the analysis of these accidents and ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

Correction letter of 12-15-2003

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the Virgil C. Summer site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcuries/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

The temperature and pressure changes during heatup and cool down are limited by curves developed using the methodology from Westinghouse Topical Report, WCAP-14040-NP-A, updated to include the requirements of the 1996 ASME Boiler and Pressure Vessel Code, Section XI, Appendix G along with ASME Code Case N-640.

1) The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3.

   a) Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

PRESSURE/TEMPERATURE LIMITS (Continued)

b) Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2) These limit lines shall be calculated periodically using methods acceptable to the NRC.

3) The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 625°F.

5) System in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The following Pages have been deleted:

B 3/4 4-8
B 3/4 4-9
B 3/4 4-10
B 3/4 4-10a
B 3/4 4-11
B 3/4 4-12
B 3/4 4-13

PRESSURE/TEMPERATURE LIMITS (Continued)

Limit curves for normal heatup and cooldown of the primary Reactor Coolant System have been calculated as described in Westinghouse Topical Report, WCAP-15102, Rev. 2, "V. C. Summer Unit 1 Heatup and Cooldown Curves for Normal Operation".

Transition temperature shifts occurring in the pressure vessel materials due to radiation exposure have been obtained directly from the reactor pressure vessel surveillance program. Charpy test specimens from Capsule W indicate that the core region lower shell plate code no. C9923-1, -2 are the limiting beltline materials for all heatup and cooldown curves to be generated. These materials exhibit limiting ART values of 107°F at 1/4T and 94°F at 3/4T at a calculated inner surface fluence of 3.84 x 10^{19} n/cm^2 at 32 EFPY.

Allowable combinations of temperature and pressure for specific temperature change rates are below and to the right of the limit lines shown on the heatup and cooldown curves. The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 3.4-2. This is in addition to other criteria which must be met before the reactor is made critical, as discussed in the following paragraphs.

The leak test limit curve shown in Figure 3.4-2 represents minimum temperature requirements at the leak test pressure specified by applicable codes. The leak test limit curve was determined by methods of the Standard Review Plan, Chapter 5.3.2 and Appendix G of the ASME Code, Section XI.

The reactor must not be made critical until pressure-temperature combinations are to the right of the criticality limit line shown in Figure 3.4-2. The criticality limit curve specifies pressure - temperature limits for core operation to provide additional margin during actual power production as specified in Appendix G to 10 CFR 50. The pressure - temperature limits for core operation (except for low power physics tests) are that the reactor vessel must be at a temperature equal to or higher than the minimum temperature required for the inservice hydrostatic test, and at least 40°F higher than the minimum permissible temperature in the corresponding pressure - temperature curve for heatup and cooldown calculated as described in this technical basis. The vertical line drawn from these points on the pressure - temperature curve, intersecting a curve 40°F higher than the pressure - temperature limit curve, constitutes the limit for core operation for the reactor vessel.

Figures 3.4-2 and 3.4-3 define limits for insuring prevention of nonductile failure.

The instrument uncertainties, effects of forced flow from the reactor coolant pumps, and the elevation effect of the pressure sensors are incorporated into the curves located in the plant operating procedures.

Although the pressurizer operates in temperature ranges above those for which there is reason for concern of non-ductile failure, operating limits are provided to assure compatibility of operation with the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two RHRSRVs or an RCS vent opening of at least 2.7 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 300°F. Either RHRSRV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures or (2) the start of an HPSI pump and its injection into a water solid RCS.

The limitation for a maximum of one charging pump to be capable of injecting into the RCS, and the Surveillance Requirement to verify at least two charging pumps are demonstrated to be INOPERABLE at least once per 31 days, while the RCS is below 300°F, provides assurance that a mass addition transient can be mitigated by a single RHR suction relief valve.

3/4.4.10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

BASES

3/4.5.1 ACCUMULATORS

The OPERABILITY of each Reactor Coolant System (RCS) accumulator ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the accumulators. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures. In addition, the borated water serves to limit the maximum power which may be reached during large secondary pipe ruptures.

The limits on accumulator volume, boron concentration and pressure ensure that the assumptions used for accumulator injection in the safety analysis are met.

The accumulator power operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these accumulator isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with an accumulator inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional accumulator which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one accumulator is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 EMERGENCY CORE COOLING SYSTEM (ECCS) SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the accumulators is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

ECCS SUBSYSTEMS (Continued)

The limitation for a maximum of one centrifugal charging pump to be OPERABLE and the Surveillance Requirement to verify all charging pumps except the required OPERABLE charging pump to be inoperable below 300°F provides assurance that a mass addition pressure transient can be relieved by the operation of a single RHR suction relief valve.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. Surveillance requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 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 either a LOCA, a steamline break or inadvertent RCS depressurization. The limits of RWST minimum volume and boron concentration ensure 1) that sufficient water is available within containment to permit recirculation cooling flow to the core, 2) that the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, Spray Additive Tank (SAT), containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1), 3) that the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area $\geq$3.0 sq. ft.) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO), 4) long term subcriticality following a steamline break assuming ARI-1 and preclude fuel failure.

The maximum allowable value for the RWST boron concentration forms the basis for determining the time (Post-LOCA) at which operator action is required to switch over the ECCS to hot leg recirculation in order to avoid precipitation of the soluble boron.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics

ECCS SUBSYSTEMS (Continued)

The limits on contained water volume and boron concentration of the RWST also ensure a pH value of between 7.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates (including those used in demonstrating a 30 day water seal) ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic test to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates is consistent with the Containment Leakage Rate Testing Program.

3/4.6.1.3 REACTOR BUILDING AIR LOCKS

The limitations on closure for the reactor building air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on reactor building internal pressure ensure that 1) the reactor building structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.5 psig and 2) the reactor building peak pressure does not exceed the design pressure of 57 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 53.5 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 53.5 psig which is less than design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on reactor building average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a steam line break accident.

3/4.6.1.6 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 53.5 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability.

The reactor building structural integrity limitations as described in the Containment Inservice Inspection Program (CISIP) ensure 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 53.5 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment, and the Type A leakage test are sufficient to demonstrate this capability. Visual and other required examinations of tendons, anchorages, and surfaces are performed periodically in accordance with plant procedures. These procedures embody applicable requirements of the 1992 Addenda of ASME Code, Section XI, Subsection IWL as set forth in 10CFR50.55a. Any degradations exceeding the CISIP acceptance criteria will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.

CONTAINMENT STRUCTURAL INTEGRITY (Continued)

In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

The tendon lift-off forces are evaluated to ensure that 1) the rate of tendon force loss is within predicted limits, and 2) a minimum required tendon force level exists in the containment. In order to assess the rate of force loss, the average lift off force for a tendon is compared with 95% of the predicted force. The predicted force is calculated by subtracting the initial, time-dependent, and other losses where applicable from the original stressing force, consistent with the recommendations of Regulatory Guide 1.35.1, Revision 3 dated July 1990.

3/4.6.1.7 REACTOR BUILDING VENTILATION SYSTEM

The 36-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 36-inch valves cannot be inadvertently opened, they are sealed closed in accordance with the Standard Review Plan 6.2.4 which includes mechanical devices to seal or lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 6 inch purge supply and exhaust isolation valves since unlike the 36 inch valves the 6 inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR 100 would not be exceeded in the event of an accident during purging operations.

Periodic leakage integrity tests for purge supply and exhaust isolation valves with resilient material seals will be performed in accordance with the Containment Leakage Rate Testing Program.

3/4.6.2. DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 REACTOR BUILDING SPRAY SYSTEM

The OPERABILITY of the reactor building spray system ensures that reactor building depressurization and cooling capability will be available in the event of a steam line break. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

The reactor building spray system and the reactor building cooling system are redundant to each other in providing post accident cooling of the reactor building atmosphere. However, the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH is added to the reactor building spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 7.5 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained solution volume limit includes an allowance for solution not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 REACTOR BUILDING COOLING SYSTEM

The OPERABILITY of the reactor building cooling system ensures that 1) the reactor building air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the reactor building spray systems during post-LOCA conditions.

The reactor building cooling system and the reactor building spray system are redundant to each other in providing post accident cooling of the reactor building atmosphere. As a result of this redundance in cooling capability, the allowable out of service time requirements for the reactor building cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the reactor building spray system have been maintained consistent with that assigned other inoperable ESF equipment since the reactor building spray system also provides a mechanism for removing iodine from the reactor building atmosphere.

The accident analysis requires the service water booster pump to be passing 4,000 gpm to both RBCU's within 86.5 seconds. This time encompasses the driving of all necessary service water valves to the correct positions, i.e., fully opened or fully closed. Reference Technical Specification Bases B 3/4.3.1 and B 3/4.3.2 for additional details.

3/4.6.3 PARTICULATE IODINE CLEANUP SYSTEM

The OPERABILITY of the containment filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the reactor building atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the reactor building atmosphere or pressurization of the reactor building and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits required by the safety analysis 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.

The opening of locked or sealed closed containment isolation valves on an intermittent basis under administrative control includes the following considerations: (1) stationing an operator, who is in constant communication with control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves.

3/4.6.5 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within the reactor building below its flammable limit during post-LOCA conditions. Either recombiner unit (or the purge system) is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water, and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA," March 1971.

The hydrogen mixing systems are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1305 psig) of its design pressure of 1185 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is sufficient to limit secondary side pressure to within 110% of design at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for the allowable THERMAL POWER restriction in Table 3.7-1.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Range Neutron Flux channels. The reactor trip setpoint reductions are based on the plant safety analysis or are conservatively derived on the following bases, implementing Westinghouse NSAL 94-01 Methodology:

$$SP = Hi\,\phi - \xi = \left(\frac{100\%}{Q}\right)\frac{w_s \cdot h_{fg} \cdot N}{K} - \xi$$

Where:

SP = Reduced Reactor Trip Setpoint [% RATED THERMAL POWER (RTP)]

Hi ϕ = Safety Analysis Power Range High Neutron Flux Trip Setpoint (% RTP)

ξ = Power Range Neutron Flux Channel Uncertainty (% RTP)

Q = RTP plus Reactor Coolant Pump Heat (Mwt)

W_s = Minimum total steam flow rate capability of the operable main steam line code safety valves on any one steam generator at the highest MSSV opening pressure including allowances for setpoint tolerance and accumulation (lb/sec)

h_{fg} = Heat of vaporization for steam at the highest main steam line code safety valve operating pressure including allowances for setpoint tolerance and accumulation (Btu/lb)

N = Number of Loops in Plant

K = Conversion factor, 947.82 $\frac{\text{(Btu/sec)}}{\text{Mwt}}$

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEM

The OPERABILITY of the emergency feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each emergency feedwater pump is capable of delivering a total feedwater flow of 380 gpm at a pressure of 1211 psig to the entrance of two out of three steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F at which point the Residual Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the RCS at HOT STANDBY conditions for 11 hours with steam discharge to the atmosphere concurrent with total loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within the reactor building in the event the steam line rupture occurs within the reactor building. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 FEEDWATER ISOLATION VALVES

The OPERABILITY of the Feedwater Isolation Valves serves to (1) limit the effects of a Steam Line rupture by minimizing the positive reactivity effects of the Reactor Coolant System Cooldown associated with the blowdown, and (2) limit the pressure rise within the reactor building in the event of a Steam Line or Feedwater Line rupture within the reactor building.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200 psig are based on the average impact valves of the steam generator material at 10°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the component cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 SERVICE WATER SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident conditions within acceptable limits.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

ULTIMATE HEAT SINK (Continued)

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants", March 1974.

3/4.7.6 CONTROL ROOM NORMAL AND EMERGENCY AIR HANDLING SYSTEM

The OPERABILITY of the control room ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criteria 19 of Appendix "A", 10 CFR 50.

3/4.7.7 SNUBBERS

All snubbers on systems required for safe shutdown/accident mitigation shall be OPERABLE. This includes safety and non-safety related snubbers on systems used to protect the code boundary and to ensure the structural integrity of these systems under dynamic loads.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip and 100 kip capacity manufactured by company "A" are of the same type. The same design mechanical snubber manufactured by company "B" for the purposes of this specification would be of a different type, as would hydraulic snubbers from either manufacturer.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Table 4.7-2 establishes three limits for determining the next visual inspection interval. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. Any inspection whose results require a shorter inspection interval will override the previous schedule.

SNUBBERS (Continued)

To provide assurance of snubber functional reliability one of two sampling and acceptance criteria methods are used:

1) functionally test 10 percent of a type of snubber with an additional 10 percent tested for each functional testing failure, or

2) functionally test a sample size and determine sample acceptance using Figure 4.7-1.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted shall be listed in Section 3.7.7 with footnotes indicating the extent of the exemptions.

3/4.7.8 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.7.9 AREA TEMPERATURE MONITORING

The area temperature limitations ensure that safety-related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of 2°F.

3/4.7.10 WATER LEVEL - SPENT FUEL POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.7.11 SPENT FUEL POOL VENTILATION SYSTEM

The limitations on the spent fuel pool ventilation system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analysis.

3/4.7.12 SPENT FUEL ASSEMBLY STORAGE

The restrictions placed on spent fuel assemblies in Region 2 of the spent fuel pool ensure K_{eff} remains less than 0.95. The minimum burnup bounds the use of Burnable Poison Rod Assemblies (BPRA), Wetted Annular Burnable Absorbers (WABA), Integral Fuel Burnable Absorbers (IFBA), and Erbia.

An axial burnup shape penalty is also included in the burnup requirement.

3/4.7.13 SPENT FUEL POOL BORON CONCENTRATION

A minimum boron concentration is required in the spent fuel pool, fuel transfer canal, or cask loading pit whenever new 4.95 W/0 fuel is being moved to ensure K_{eff} remains less than 0.95 during this normal condition of fuel movement.

The minimum boron concentration in the spent fuel pool, fuel transfer canal, or cask loading pit also is sufficient to maintain K_{eff} less than 0.95 for postulated accident condition consisting of a dropped or a mispositioned fuel assembly. This requirement is a direct result of the analysis performed for pool criticality during evolutions performed while the Spent Fuel Pool is isolated from the reactor cavity and is due to geometry, materials and poisons being different in the spent fuel pool than those in the reactor. During periods of direct communication between the pool and the reactor, Specification 3.9.1 shall be followed when the refueling cavity is filled and the transfer canal blind flange is removed.

Sampling to determine boron concentration is required only for those specific areas where fuel is being moved, e.g. in the spent fuel pool, in the fuel transfer canal, or in the cask loading area.

3/4.8.1, 3/4.8.2 AND 3/4.8.3 A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

If it can be determined that the cause of the inoperable diesel generator does not exist on the OPERABLE diesel generator, then Surveillance 4.8.1.1.2.a.3 does not have to be performed. If the cause of the initial inoperable diesel generator cannot be confirmed not to exist on the redundant diesel generator, performance of Surveillance Requirement 4.8.1.1.2.a.3 suffices to provide assurance of continued OPERABILITY of that diesel generator. This allows for reduced start testing of the diesel generators, which has been shown to be a factor in engine degradation.

In the event that the inoperable diesel generator is restored to OPERABLE status prior to completing either the evaluation of cause or performing the surveillance requirement, the CER program will continue to evaluate the common cause possibility. This continued evaluation, however, is no longer under the 24 hour constraint imposed by the action statement. According to Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," 24 hours is reasonable to confirm that the OPERABLE diesel generator is not affected by the same problem as the inoperable diesel generator.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

A.C. SOURCES. D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979, as modified by the NRC's review and approval of South Carolina Electric & Gas Company's June 10, 1985, December 6, 1985, and November 10, 2000 amendment requests.

The Surveillance Requirement that assures the diesel generator is capable of performing its design function follows the guidance of NUREG 1366 and NUREG 1431, Rev 2. The surveillance tests the capability of the diesel generator to start and close its breaker in the required 10 seconds to support the accident analysis, and carry the required electrical load while maintaining the voltage and frequency limits necessary to assure OPERABILITY of the loads.

In addition to the Surveillance Requirements, the time for the diesel generator to reach steady state operation, unless the modified start method is utilized, is periodically monitored and the trend evaluated to identify degradation of the governor and voltage regulator performance.

The fuel storage system minimum volume of fuel to demonstrate operability of the diesel generators was based on fuel consumption determined from the development of time dependent loads following a design basis accident and a loss of off-site power utilizing FSAR Table 8.3-3 for seven days.

All safety-related portions of the VCSNS diesel engine fuel oil storage and transfer system, are Seismic Category I, Safety Class 2b, and designed to ANSI Standard N195-1976 with the provision listed below:

> VCSNS will maintain at least 2% margin above the minimum calculated seven day required volume during Modes 1-4. This is an exception to ANSI N195-1976, "Fuel Oil Systems for Standy Diesel Generators," Section 5.4, during Modes 1-4. EDG fuel replenishment is available from multiple sources, including off-site suppliers, on-site non safety storage in the Auxiliary Boiler Fuel Tank, and the ability to provide fuel from the opposite train EDG Fuel Oil Storage Tank via the fuel oil and transfer system cross-tie.
>
> The 10% fuel margin as recommended in Regulatory Guide 1.137, Revision 1, "Fuel-Oil Systems for Standby Diesel Generators," position C.1.c.(2) will be met during Modes 5 and 6.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1987, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage and float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

A.C. SOURCES. D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Table 4.8-2 specifies the normal limits far each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

Containment electrical penetrations and penetration conductors are protected by either deenergizing circuits not required during reactor operation or by demonstrating the OPERABILITY of primary and backup overcurrent protection circuit breakers during periodic surveillance.

The surveillance requirements applicable to lower voltage circuit breakers provide assurance of breaker reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker. Each manufacturer's molded case and metal case circuit breakers are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers are tested. If a wide variety exists within any manufacturer's brand of circuit breakers, it is necessary to divide that manufacturer's breakers into groups and treat each group as a separate type of breaker for surveillance purposes.

The surveillance requirements of the circuit breakers for non-Class 1E cables located in trays which do not have cable tray covers and which provide protection for cables that, if faulted, could cause failure in both adjacent, redundant Class 1E cables ensures that the integrity of Class 1E cables is not compromised by the failure of protection devices to operate in the non-Class 1E cables.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses. The value of 0.95 or less for K_{eff} includes a 1 percent delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 2000 ppm or greater includes a conservative uncertainty allowance of 50 ppm boron. Valves in the reactor makeup system are required to be closed to minimize the possibility of a boron dilution accident.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum time of 72 hours for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short-lived fission products. The minimum decay time of 72 hours is consistent with the assumptions used in the accident analysis.

The tabulated hold times associated with Component Cooling Water (CCW) temperature ensure that the spent fuel heat load is reduced sufficiently to allow the spent fuel pool cooling system to maintain the bulk pool temperature below 170°F. These hold times ensure that adequate cooling is provided to the Spent Fuel Pool under the highest possible heat load conditions. The hold times are based on the performance of the cooling system, which is dependent upon CCW temperature and recognizes that the spent fuel pool cooling system is capable of increased flow rates up to 2400 gpm during single loop operation. This higher flow rate may be required when only a single cooling loop is operable during a refueling outage.

The CCW temperature limits defined in Figure 3.9-1 are adjusted for uncertainty in the implementing procedure.

3/4.9.4 REACTOR BUILDING PENETRATIONS

The requirements on reactor building penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of reactor building pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity conditions during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the manipulator cranes ensure that: 1) manipulator cranes will be used for movement of control rods and fuel assemblies, 2) each crane has sufficient load capacity to lift a control rod and fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and 2) sufficient coolant circulation is maintained thru the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two RHR loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange ensures that a single failure of the operating RHR loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and at least 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling. Thus, in the event of a failure of the operating RHR loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.8 REACTOR BUILDING PURGE SUPPLY AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the reactor building vent and purge penetrations will be automatically isolated upon detection of high radiation levels within the reactor building. The OPERABILITY of this system is required to restrict the release of radioactive material from the reactor building atmosphere to the environment.

3/4.9.9 WATER LEVEL - REACTOR VESSEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.10 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 1) measure control rod worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PHYSICS TESTS

This special test exception permits PHYSICS TESTS to be performed at less than or equal to 5% of RATED THERMAL POWER with the RCS T_{avg} slightly lower than normally allowed so that the fundamental nuclear characteristics of the reactor core and related instrumentation can be verified. In order for various characteristics to be accurately measured, it is at times necessary to operate outside the normal restrictions of these Technical Specifications. For instance, to measure the moderator temperature coefficient at BOL, it is necessary to position the various control rods at heights which may not normally be allowed by Specification 3.1.3.6 which in turn may cause the RCS T_{avg} to fall slightly below the minimum temperature of Specification 3.1.1.4.

3/4.10.4 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under no flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.5 POSITION INDICATION SYSTEM-SHUTDOWN

This special test exception permits the position indication systems to be inoperable during rod drop time measurements. The exception is required since the data necessary to determine the rod drop time is derived from the induced voltage in the position indicator coils as the rod is dropped. This induced voltage is small compared to the normal voltage and, therefore, can not be observed if the position indication systems remain OPERABLE.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 Deleted by Amendment 104.
3/4.11.1.2 Deleted by Amendment 104.
3/4.11.1.3 Deleted by Amendment 104.

3/4.11.1.4 LIQUID HOLDUP TANKS

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.

3/4.11.1.5 SETTLING PONDS

The inventory limits of the settling ponds (SP) are based on limiting the consequences of an uncontrolled release of the pond inventory. The expression in Specification 3.11.1.5 assumes the pond inventory is uniformly mixed, that the pond is located in an unrestricted area as defined in 10 CFR 20, and that the concentration limit in Note 4 to Appendix B of 10 CFR 20 applies.

The density of wet, drained resin is approximately the same as water (bulk density of about 58 pounds per cubic foot); and the absorption characteristics for gamma radiation are essentially that of water. Therefore, direct comparison of radionuclide specific activity in wet, drained resin to the appropriate concentration in 10 CFR 20, Appendix B, Table 2, Column 2, ensures that the limit of Specification 3.11.1.5 will not be exceeded.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 Deleted by Amendment 104.
3/4.11.2.2 Deleted by Amendment 104.
3/4.11.2.3 Deleted by Amendment 104.
3/4.11.2.4 Deleted by Amendment 104.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen to reduce the concentration below the flammability limits. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure".

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-3.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 REACTOR BUILDING

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, pre-stressed, post-tensioned reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 126 feet.

b. Nominal inside height = 187 feet.

c. Minimum thickness of concrete walls = 4 feet.

d. Minimum thickness of concrete roof = 3 feet.

e. Minimum thickness of concrete floor pad = 4 feet.

f. Nominal thickness of steel liner = 0.25 inches.

g. Net free volume = 1.842×10^6 cubic feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 57 psig and a temperature of 283°F.

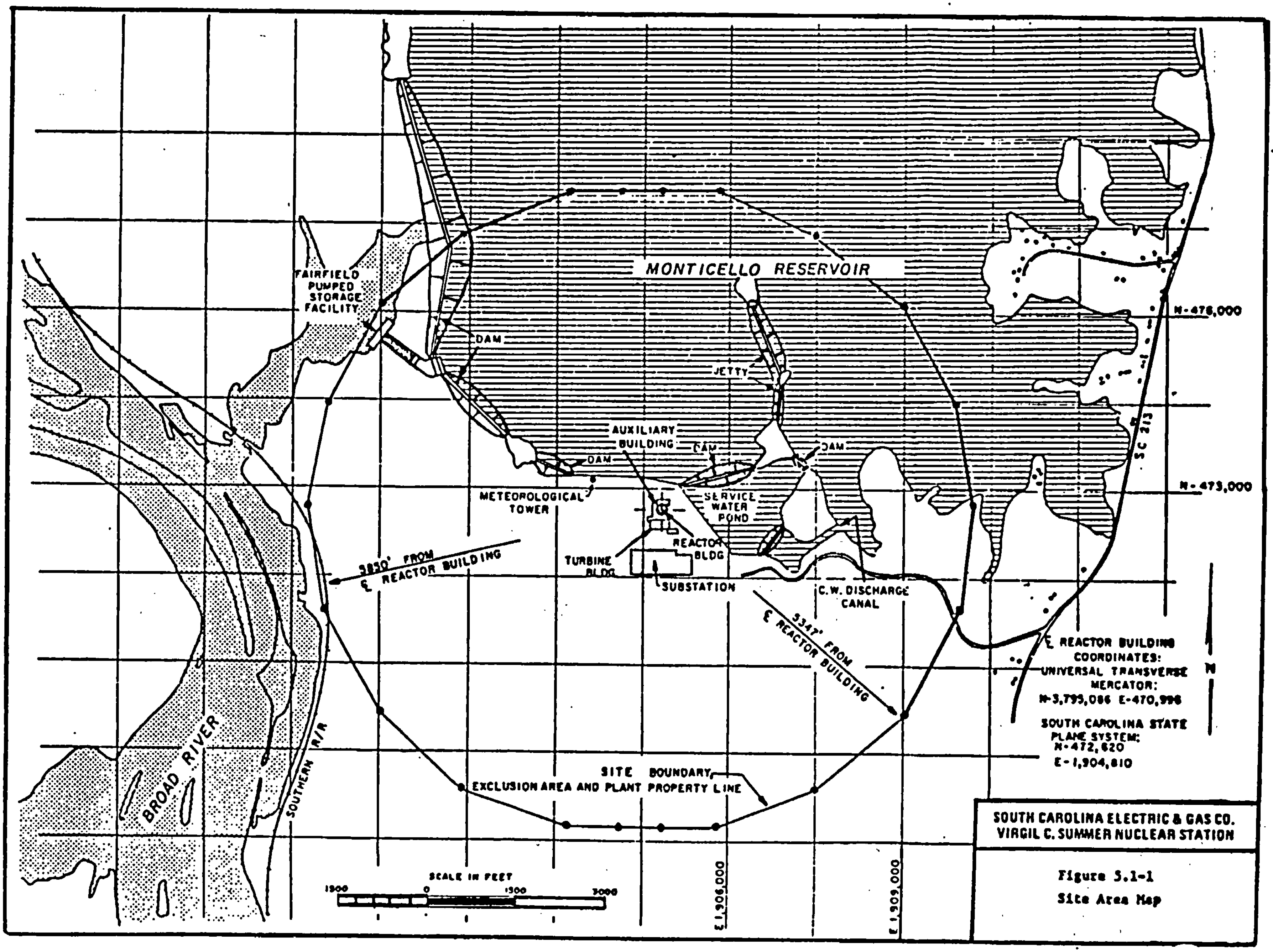


Figure 5.1-1
Site Area Map

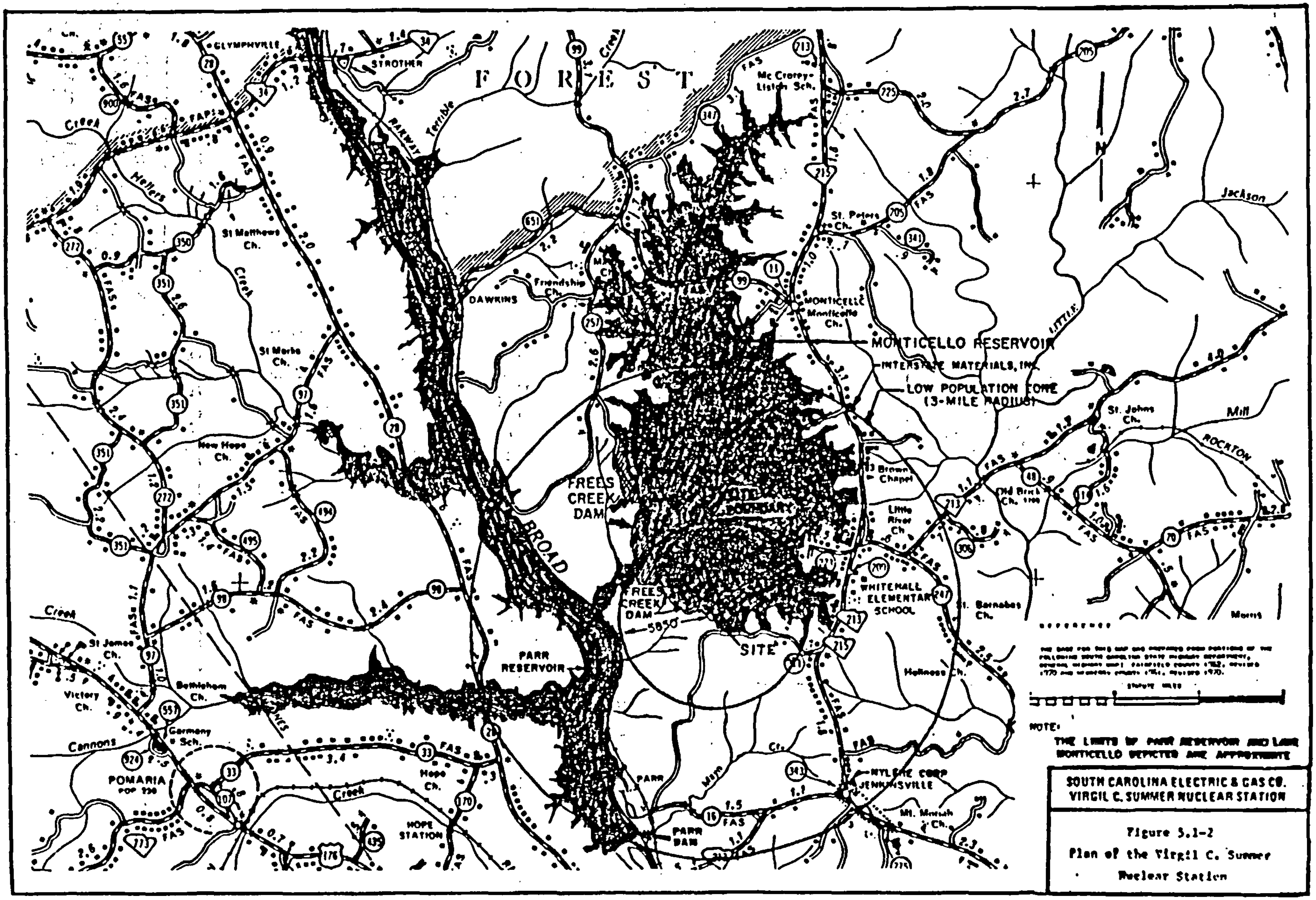
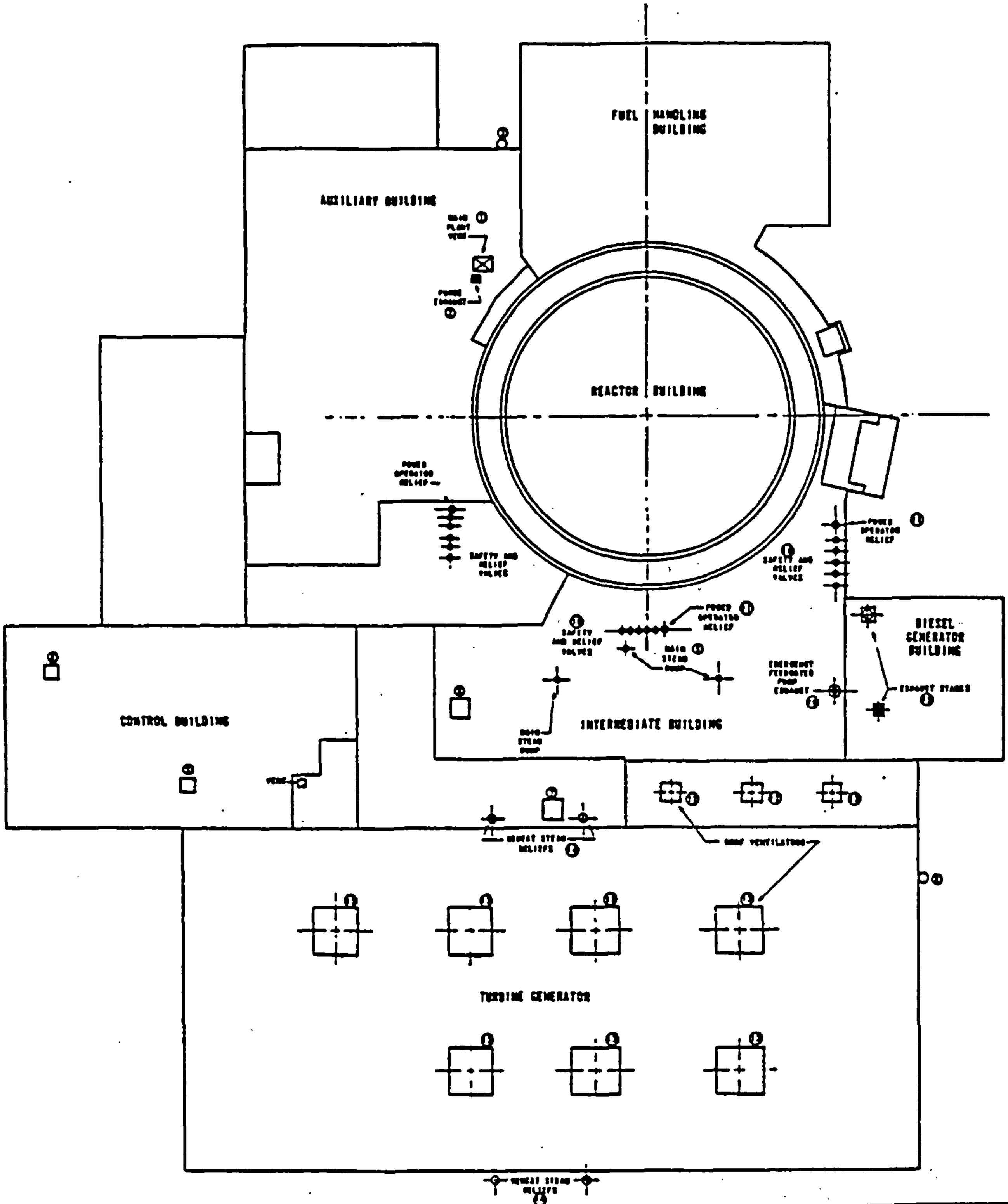


Figure 5.1-2

Plan of the Virgil C. Summer Nuclear Station

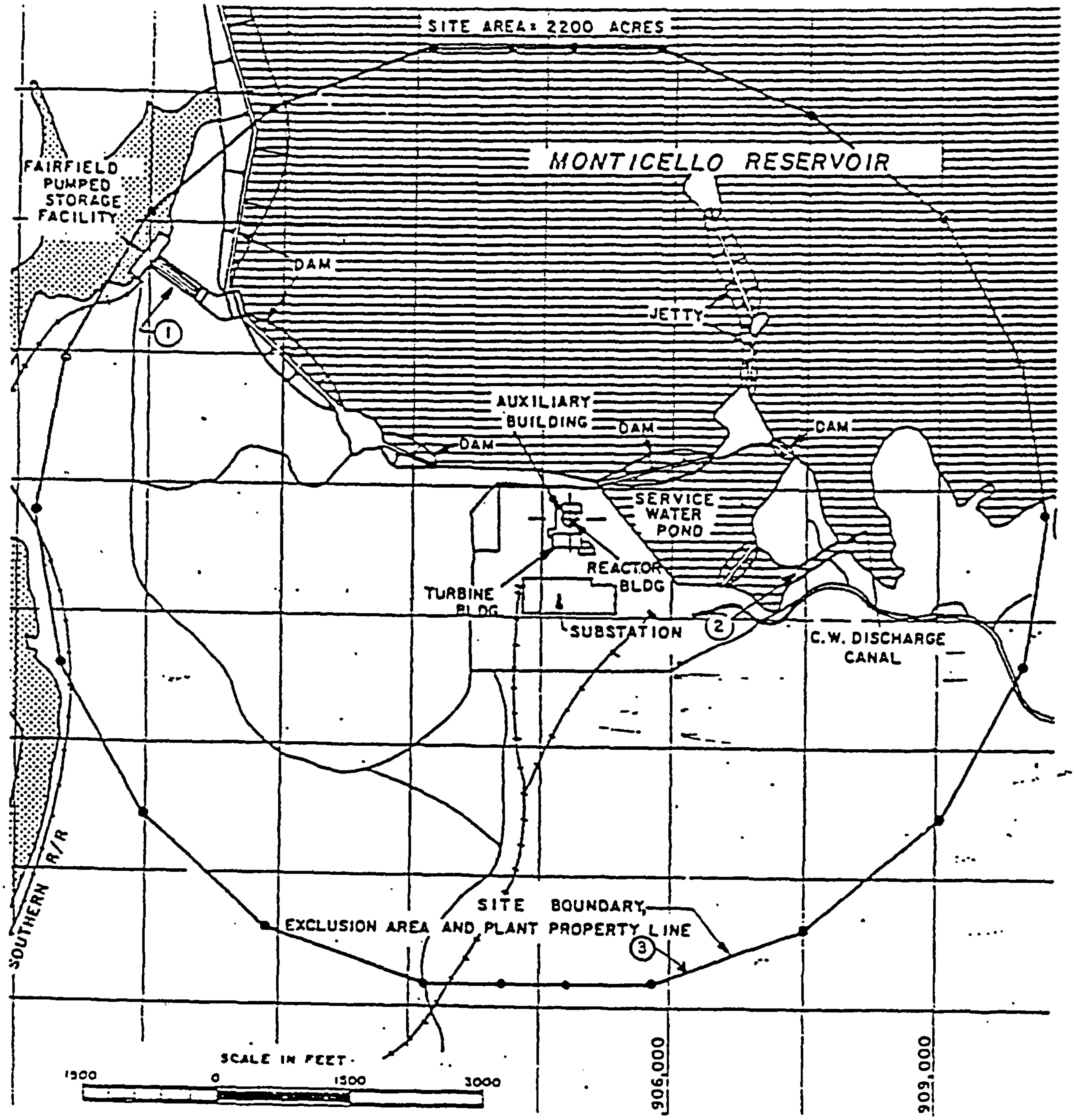


NOTE: See Figure 5.1-4 for site boundary for gaseous effluents

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Potentially Radioactive Gaseous Waste Release Points

FIGURE 5.1-3



LIQUID RELEASES:

1. FAIRFIELD PUMPED STORAGE FACILITY PENSTOCKS
   (A) LIQUID WASTE PROCESSING SYSTEM
   (B) PROCESSED STEAM GENERATOR BLOWDOWN
2. CIRCULATING WATER DISCHARGE CANAL
   (A) UNPROCESSED STEAM GENERATOR BLOWDOWN
   (B) TURBINE BUILDING FLOOR DRAINS

GASEOUS RELEASES:

3. SITE BOUNDARY FOR GASEOUS RELEASES

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VIRGIL C. SUMMER NUCLEAR STATION

Location of Liquid Release Points

FIGURE 5.1-4

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The core shall contain 157 fuel assemblies. Each fuel assembly shall consist of 264 Zircaloy-4 or ZIRLO$^{(TM)}$ clad fuel rods with an initial composition of uranium dioxide with a maximum nominal enrichment of 4.95 weight percent U-235 as fuel material. Limited substitutions of Zircaloy-4, ZIRLO$^{(TM)}$ and/or stainless steel filler rods for fuel rods, if justified by a cycle specific reload analysis using an NRC-approved methodology, may be used. Fuel assembly configurations shall be limited to those designs that have been analyzed with applicable NRC staff-approved codes and methods, and shown by tests or cycle-specific reload analyses to comply with all fuel safety design bases. Reload fuel shall contain sufficient integral fuel burnable absorbers such that the requirements of Specifications 5.6.1.1a.2 and 5.6.1.2.b are met. A limited number of lead test assemblies that have not completed representative testing may be placed in non-limiting core locations.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 680°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 9914 ± 100 cubic feet at an indicated T_{avg} of 587.4°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks consist of 1712 individual storage cells. The cells are grouped into two regions, which are determined based on storage cell spacing as defined below. The spent fuel storage racks are designed, and shall be maintained, with a K_{eff} less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases. This is ensured by maintaining the following for each region:

a. REGION 1 - designated for storage of fresh fuel assemblies and fuel assemblies with a cumulative burnup less than the required cumulative burnup for storage in Region 2.

   1. A nominal 10.867 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A maximum nominal initial enrichment of 4.95 weight percent U-235.

b. REGION 2 - designated for storage of discharged fuel assemblies.

   1. A nominal 9.07 inch center-to-center distance between fuel assemblies placed in the storage rack.

   2. A cumulative burnup with the acceptable domain defined by Figure 3.7-1.

5.6.1.2 The new fuel storage racks consist of 60 individual cells, each of which accommodates a single assembly. The new fuel pit storage racks are designed and shall be maintained with a K_{eff} less than or equal to 0.95 when flooded with unborated water and less than or equal to 0.98 for low density optimum moderation conditions, including conservative allowances for uncertainties and biases. This is ensured by maintaining:

a. A nominal 21 inch center-to-center distance between new fuel assemblies placed in the storage rack.

b. A nominal enrichment of 5.0 weight percent U-235.

DRAINAGE

5.6.2 The spent fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 460'3".

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 1712 fuel assemblies, with 200 assemblies in Region 1 and 1512 assemblies in Region 2.

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.

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TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/hr and 200 cooldown cycles at < 100°F/hr.	Heatup cycle - T_{avg} from at ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ 550°F to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/hr.	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.
	400 reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	10 inadvertent auxiliary spray actuation cycles.	Spray water temperature differential > 320°F.
	50 leak tests.	Pressurized to ≥ 2485 psig.
	5 hydrostatic pressure tests.	Pressurized to ≥ 3107 psig.
	200 large stepload decrease with steam dump.	Load decreases of more than 10% RATED THERMAL POWER occurring in 1 minute or less.
Secondary System	1 steam line break.	Break in a > 6 inch steam line.
	5 hydrostatic pressure tests.	Pressurized to ≥ 1350 psig.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The General Manager, Nuclear Plant Operations shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Shift Supervisor shall be responsible for unit operations. A management directive to this effect, signed by the Vice President, Nuclear Operations, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

6.2.1 OFFSITE AND ONSITE ORGANIZATIONS

Offsite and Onsite organizations shall be established for unit operation and corporate management, respectively. The offsite and onsite organizations 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 levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts, functional descriptions of department responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. The organizational charts will be documented in the FSAR and updated in accordance with 10 CFR 50.71(e).

b. The General Manager, Nuclear Plant Operations, shall be responsible for overall unit safe operation and shall have control over onsite activities necessary for safe operation and maintenance of the plant.

c. The Vice President, Nuclear Operations, shall have corporate responsibility 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.

d. The individuals who train the operating staff and those who carry out the health physics and quality assurance functions may report to the appropriate onsite manager; however, they shall have sufficient organizational freedom to ensure their independence from operating pressures.

6.2.2 UNIT STAFF

a. Each on-duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one Licensed Senior Reactor Operator shall be in the Control Room.

c. A health physics technician# shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

Adequate shift coverage shall be maintained without routine heavy use of overtime. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modifications, on a temporary basis, the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any seven day period, all excluding shift turnover time.

c. A break of at least eight hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the General Manager, Nuclear Plant Operations, or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Except during extended shutdown periods, controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the General Manager, Nuclear Plant Operations, or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

#The health physics technician 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.

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

SUMMER UNIT 1

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3, & 4	MODES 5 & 6
SS	1	1
CRF	1	None
RO	2	1
AO	2	1
STA	1	None

SS - Shift Supervisor with a Senior Reactor Operators License on Unit 1
CRF - Control Room Supervisor with a Senior Reactor Operators License on Unit 1
RO - Individual with a Reactor Operators License on Unit 1
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Control Room Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid RO or SRO license shall be designated to assume the Control Room command function.

6.2.3 NOT USED

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 NOT USED

6.5 REVIEW AND AUDIT

6.5.1 PLANT SAFETY REVIEW COMMITTEE (PSRC)

FUNCTION

6.5.1.1 The PSRC shall function to advise the General Manager, Nuclear Plant, Operations on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Safety Review Committee (PSRC) shall be composed of a chairman and five to ten members of the Virgil C. Summer Nuclear Station management staff at the manager level or above. These positions will be designated by the General Manager, Nuclear Plant Operations in a Station Administrative Procedure. At a minimum, representatives from Operations, Maintenance, Chemistry and Health Physics, and Engineering Services shall be appointed to the PSRC.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PSRC Chairman to serve on a temporary basis; however, alternates, including the chairman's alternate, who are participating as voting members in the PSRC shall hold a minority vote in all PSRC activities.

MEETING FREQUENCY

6.5.1.4 The PSRC shall meet at least once per calendar month and as convened by the PSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the PSRC necessary for the performance of the PSRC responsibility and authority provision of these Technical Specifications shall consist of the Chairman or his designated alternate and a majority of the PSRC appointed members including their alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Safety Review Committee shall review:

a. Station administrative procedures and changes thereto,

b. The safety evaluations for 1) procedures, 2) changes to procedures, equipment or systems, and 3) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question and all programs required by Specification 6.8 and changes thereto.

c. Proposed procedures and changes to procedures, equipment or systems which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed tests or experiments which may involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

e. Proposed changes to Technical Specifications or the Operating License.

f. Reports of violations of codes, regulations, orders, Technical Specifications, or Operating License requirements having nuclear safety significance or reports of abnormal degradation of systems designed to contain radioactive material.

g. Reports of significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

h. Review of all REPORTABLE EVENTS.

i. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

j. The plant Security Plan and changes thereto.

k. The Emergency Plan and changes thereto.

l. Items which may constitute a potential nuclear safety hazard as identified during review of facility operations.

m. Investigations or analyses of special subjects as requested by the Chairman of the Nuclear Safety Review Committee.

n. The unexpected offsite release of radioactive material and the report as described in 10 CFR 50.73.

o. Changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL.

p. The plant Fire Protection Program and revisions there to.

AUTHORITY

6.5.1.7 The Plant Safety Review Committee shall:

a. Recommend in writing to the General Manager, Nuclear Plant Operations, approval or disapproval of items considered under 6.5.1.6a, c, d, e, j, and k above.

b. Render determinations in writing to the General Manager, Nuclear Plant Operations, with regard to whether or not each item considered under 6.5.1.6a, c, and d above constitutes an unreviewed safety question.

c. Make recommendations in writing to the General Manager, Nuclear Plant Operations, that actions reviewed under 6.5.1.6(b) above did not constitute an unreviewed safety question.

d. Provide written notification within 24 hours to the Vice President, Nuclear Operations and the Nuclear Safety Review Committee of disagreement between the PSRC and the General Manager, Nuclear Plant Operations however, the General Manager, Nuclear Plant Operations shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

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RECORDS

6.5.1.8 The Plant Safety Review Committee shall maintain written minutes of each PSRC meeting that, at a minimum, document the results of all PSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Vice President, Nuclear Operations, and the Chairman of the Nuclear Safety Review Committee.

6.5.2 NUCLEAR SAFETY REVIEW COMMITTEE (NSRC)

FUNCTION

6.5.2.1 The Nuclear Safety Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemsitry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 NSRC shall consist of a Chairman and four or more other members appointed by the Vice President, Nuclear Operations. No more than a minority of the members of the NSRC shall have line responsibility for the operation of the unit.

The NSRC members shall hold a Bachelor's degree in an engineering or physical science field or equivalent experience and a minimum of five years of technical experience of which a minimum of three years shall be in one or more of the disciplines of 6.5.2.1a through h. In the aggregate, the membership of the committee shall provide specific practical experience in the majority of the disciplines of 6.5.2.1a through h.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the Vice President, Nuclear Operations; however, no more than two alternates shall participate as voting members in NSRC activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the NSRC Chairman to provide expert advice to the NSRC.

MEETING FREQUENCY

6.5.2.5 The NSRC shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of the NSRC necessary for the performance of the NSRC review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least 3 NSRC members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The NSRC shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems, and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. 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.

i. Reports and meetings minutes of the Plant Safety Review Committee.

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AUDITS

6.5.2.8 The NSRC shall have cognizance of the audits listed below. Audits may be performed by using established SCE&G groups such as QA or by outside groups as determined by the NSRC. Audit reports or summaries will be the basis for NSRC action:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions.

b. The performance, training and qualifications of the entire unit staff.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50.

e. The Emergency Plan and implementing procedures.

f. The Security Plan and implementing procedures.

g. Any other area of unit operation considered appropriate by the NSRC or the Vice President, Nuclear Operations.

h. The Fire Protection Program and implementing procedures.

i. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or a qualified outside firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

k. The radiological environmental monitoring program and the results thereof, including the performance of activities required by the quality assurance program per R.G. 4.15 Rev. 1, February 1979.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidification of radioactive wastes.

AUTHORITY

6.5.2.9 The NSRC shall report to and advise the Vice President, Nuclear Operations, on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 Records of NSRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NSRC meeting shall be prepared, approved and forwarded to the Vice President, Nuclear Operations, within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Vice President, Nuclear Operations, within 14 days following completion of the review.

c. Audit summary reports encompassed by Section 6.5.2.8 above, shall be forwarded to the NSRC and the Vice President, Nuclear Operations. Full audits shall be forwarded to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.5.3 TECHNICAL REVIEW AND CONTROL

ACTIVITIES

6.5.3.1 Activities which affect nuclear safety shall be conducted as follows:

a. Procedures required by Technical Specification 6.8 and other procedures which affect plant nuclear safety, and changes thereto, shall be prepared, reviewed and approved. Each such procedure or procedure change shall be reviewed by an individual/group other than the individual/group which prepared the procedure or procedure change, but who may be from the same organization as the individual/group which prepared the procedure or procedure change. Procedures other than Administrative Procedures will be approved as delineated in writing by the General Manager, Nuclear Plant Operations. The General Manager, Nuclear Plant Operations will approve administrative procedures, security implementing procedures, and emergency plan implementing procedures. Temporary approval to procedures which clearly do not change the intent of the approved procedures can be made by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License. For changes to procedures which may involve a change in intent of the approved procedures, the person authorized above to approve the procedures shall approve the change.

b. Proposed changes or modifications to plant nuclear safety-related structures, systems and components shall be reviewed as designated by the General Manager, Nuclear Plant Operations. Each such modification shall be designed as authorized by Engineering Services and shall be reviewed by an individual/group other than the individual/group which designed the modification, but who may be from the same organization as the individual/group which designed the modifications. Implementation of modifications to plant nuclear safety-related structures, systems and components shall be concurred in by the General Manager, Nuclear Plant Operations.

c. Proposed tests and experiments which affect plant nuclear safety and are not addressed in the Final Safety Analysis Report shall be reviewed by an individual/group other than the individual/group which prepared the proposed test or experiment.

d. Events reportable pursuant to the Technical Specification 6.9 and violations of Technical Specifications shall be investigated and a report prepared which evaluates the event and which provides recommendations to prevent recurrence. Such report shall be approved by the General Manager, Nuclear Plant Operations and forwarded to the Chairman of the Nuclear Safety Review Committee.

e. Individuals responsible for reviews performed in accordance with 6.5.3.1.a, 6.5.3.1.b, 6.5.3.1.c and 6.5.3.1.d shall be members of the plant staff that meet or exceed the qualification requirements of Section 4 of ANSI 18.1, 1971, as previously designated by the General Manager, Nuclear Plant Operations. Each such review shall include a determination of whether or not additional, cross-disciplinary, review is necessary. If deemed necessary, such review shall be performed by the review personnel of the appropriate discipline.

f. Each review will include a determination of whether or not an unreviewed safety question is involved.

RECORDS

6.5.3.2 Records of the above activities shall be provided to the General Manager, Nuclear Plant Operations, PSRC and/or NSRC as necessary for required reviews.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and/or 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 PSRC and the results of this review shall be submitted to the NSRC and the Vice President, Nuclear Operations.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Vice President, Nuclear Operations and the NSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSRC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the NSRC and the Vice President, Nuclear Operations within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.
b. Refueling operations.
c. Surveillance and test activities of safety-related equipment.
d. Security Plan.
e. Emergency Plan.
f. Fire Protection Program.
g. PROCESS CONTROL PROGRAM.
h. OFFSITE DOSE CALCULATION MANUAL.
i. Effluent and environmental monitoring program using the guidance in Regulatory Guide 4.15, Revision 1, February 1979.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed prior to implementation as set forth in 6.5 above.

6.8.3 NOT USED.

6.8.4 The following programs shall be established, implemented and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include the chemical and volume control, letdown, safety injection, residual heat removal, nuclear sampling, liquid radwaste handling, gas radwaste handling and reactor building spray system. The program shall include the following:

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

1) Training of personnel,
2) Procedures for monitoring, and
3) Provisions for maintenance of sampling and analysis equipment.

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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, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

4) Procedures for the recording and management of data,

5) Procedures defining corrective actions for all off-control point chemistry conditions,

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. Not Used

e. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to members of the public from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determinations in accordance with the methodology in the ODCM;

2) Limitations on the concentration of radioactive material released in liquid effluents to unrestricted areas conforming to 10 times the concentration values in 10 CFR Part 20, Appendix B, Table 2, Column 2;

e. Radioactive Effluent Controls Program (Continued)

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM;

4) Limitations on the annual and quarterly doses or dose commitment to a member of the public from radioactive materials in liquid effluents released to unrestricted areas conforming to Appendix I to 10 CFR Part 50;

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days;

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases or radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual or dose commitment conforming to Appendix I to 10 CFR Part 50;

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents from the site to areas at or beyond the site boundary shall be limited to the following:

(a) For noble gases: Less than or equal to a dose rate of 500 mrem/yr to the total body and less than or equal to a dose rate of 3000 mrem/yr to the skin; and

(b) For Iodine-131, Iodine-133, tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to a dose rate of 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

9) Limitations on the annual and quarterly doses to a member of the public from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate from with half-lives greater than 8 days in gaseous effluents released to areas beyond the site boundary conforming to Appendix I to 10 CFR Part 50;

10) Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

f. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measures of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM;

2) A Land Use Census to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the monitoring program are made if required by the results of the census; and

3) Participation in an Inter-laboratory Comparison Program to ensure that independent checks on the precision and accuracy of measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

g. Containment Leakage Rate Testing Program

A program shall be established to implement leakage rate testing of the containment system as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program", dated September 1995; NEI 94-01, "Industry Guideline for Performance-Based Option of 10 CFR 50, Appendix J", Revision 0; ANSI/ANS-56.8-1994, "Containment System Leakage Testing Requirements"; as modified by approved exceptions.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 45.1 psig.

The maximum allowable containment leakage rate, L_a, at P_a, is 0.20 percent by weight of the containment air per 24 hours.

Leakage rate acceptance criteria are:

1) Containment overall leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C tests, and $\leq 0.75\ L_a$ for Type A tests;

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g. Containment Leakage Rate Testing Program (Continued)

2) Air lock testing acceptance criteria are:

a. Overall air lock leakage rate is ≤ 0.10 L_a when tested at ≥ P_a.

b. For each door, leakage rate is ≤ 0.01 L_a when pressurized to ≥ 8.0 psig for at least 3 minutes.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

h. Containment Inservice Inspection Program

This program provides controls for monitoring containment vessel structural integrity including routine inspections and tests to identify degradation and corrective actions if degradation is found. The Containment Inservice Inspection Program, inspection frequencies and acceptance criteria shall be in accordance with 10CFR50.55a as modified by approved exemptions. Predicted lift-off forces shall be determined consistent with the recommendations of Regulatory Guide 1.35.1, Revision 3 dated July 1990.

Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the Containment Inservice Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion of necessary repairs, and the extent, nature, and frequency of additional examinations.

In addition, any significant degradation which seriously challenges containment operability found during the inspection shall be reported to the NRC in accordance with Technical Specification 6.9.2 within 30 days. The report shall include the description of degradation, operability determination, root cause determination, and corrective actions taken.

i. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases shall be made under appropriate administrative control and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a) A change in the TS incorporated in the license or

   b) A change to the updated FSAR or bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to insure that the Bases are maintained consistent with the FSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.i.2.b above shall be reviewed and approved prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

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 Regional Administrator Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORT

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 31 of each year. The initial report shall be submitted prior to March 31 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated collective deep dose equivalent (reported in person-rem) according to work and job functions, [1/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

[1/] This tabulation supplements the requirements of §20.2206 of 10 CFR Part 20.

This report shall also include the results of specific activity analysis 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; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT

6.9.1.6 The annual radiological environmental operating report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year.

The report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

6.9.1.7 Not used.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT

6.9.1.8 Annual radioactive effluent release report covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year.

The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

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6.9.1.9 Not used.

MONTHLY OPERATING REPORT

6.9.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORV's or safety valves, shall be submitted on a monthly basis to the Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

A report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted as set forth in 6.5 above.

CORE OPERATING LIMITS REPORT

6.9.1.11 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT prior to each reload cycle, or prior to any remaining portion of a reload cycle, for the following:

a. Moderator Temperature Coefficient BOL and EOL Limits and 300 ppm surveillance limit for Specification 3/4.1.1.3,

b. Shutdown Rod Insertion Limit for Specification 3/4.1.3.5,

c. Control Rod Insertion Limits for Specification 3/4.1.3.6,

d. Axial Flux Difference Limits, target band, and APL^{ND} for Specification 3/4.2.1,

e. Heat Flux Hot Channel Factor, F_Q^{RTP}, K(z), W(z), APL^{ND}, $W(z)_{BL}$, and $F_Q(z)$ manufacturing/measurement uncertainties for Specification 3/4.2.2,

f. Nuclear Enthalpy Rise Hot Channel Factor, $F_{\Delta H}^{RTP}$, Power Factor Multiplier, $PF_{\Delta H}$, and $F_{\Delta H}^N$ measurement uncertainties limits for Specification 3/4.2.3.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

a. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY," July 1985 (W Proprietary).

CORE OPERATING LIMITS REPORT (Continued)

(Methodology for Specification 3.1.1.3 - Moderator Temperature Coefficient, 3.1.3.5 - Shutdown Rod Insertion Limit, 3.1.3.6 - Control Rod Insertion Limit, 3.2.1 - Axial Flux Difference, 3.2.2 - Heat Flux Hot Channel Factor, and 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor.)

b. WCAP-10216-P-A, Rev. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION", February 1994 (W Proprietary).

(Methodology for Specifications 3.2.1 - Axial Flux Difference (Relaxed Axial Offset Control) and 3.2.2 - Heat Flux Hot Channel Factor (F_Q Methodology for W(z) surveillance requirements)).

c. WCAP-10266-P-A, Rev. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE", March 1987; Including Addendum 2-A, "BASH METHODOLOGY IMPROVEMENTS AND RELIABILITY ENHANCEMENTS", May 1988, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor).

d. WCAP-12472-P-A, "BEACON CORE MONITORING AND OPERATIONS SUPPORT SYSTEM", August 1994, (W Proprietary).

(Methodology for Specification 3.2.2 - Heat Flux Hot Channel Factor, 3.2.3 - RCS Flow Rate and Nuclear Enthalpy Rise Hot Channel Factor, and 3.2.4 - Quadrant Power Tilt Ratio).

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements there to shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License.

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities as specified in the NRC's approved SCE&G position on Regulatory Guide 1.88, Rev. 2, October 1976.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the PSRC and the NSRC.

l. Records of the service lives of all hydraulic and mechanical snubbers defined in Section 3.7.7 including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Records of analysis required by the radiological environmental monitoring program.

o. Records of reviews performed for changes made to the Offsite Dose Calculation Manual and the Process Control Program.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREAS

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr* but less than 1000 mrem/hr* shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP). Health Physics personnel or individuals escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned duties, provided they otherwise comply with approved radiation protection procedures for entry into 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.

* Measurement made at 30 cm (12 in.) from the radiation source or from any surface penetrated by the radiation.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the RWP.

6.12.2 In addition to the requirements of 6.12.1, areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the duty Shift Supervisor and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area. The maximum allowable stay time for individuals in that area shall be established prior to entry. In lieu of the stay time specification of the RWP, direct or remote continuous surveillance (such as closed circuit TV cameras) shall be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

For individual areas accessible to personnel with radiation levels greater than 1000 mrem/hr* but less than 500 rads/hr** that are located within larger areas (such as PWR containment) where no enclosure can be reasonably constructed around the individual areas, then those areas 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 by the Commission prior to implementation.

6.13.2 Changes to the PCP:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.o. This documentation shall contain:

1) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change (s); and

2) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

b. Shall become effective after review and acceptance by the PSRC and approval of the General Manager, Nuclear Plant Operations.

* Measurement made at 30 cm (12 in.) from the radiation source or from any surface penetrated by the radiation.

** Measurement made at 1 meter from the radiation source or from any surface penetrated by the radiation.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Changes to the ODCM:

a. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2.o. This documentation shall contain:

   1) Sufficient information to support the change together with appropriate analyses or evaluations justifying the change(s); and

   2) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent dose or setpoint calculations.

b. Shall become effective after review and acceptance by the PSRC and the approval of the General Manager, Nuclear Plant Operations.

c. Shall be submitted to the Commission in the form of a complete legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.

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APPENDIX B

TO RENEWED FACILITY LICENSE NO. NPF-12

FOR

VIRGIL C. SUMMER NUCLEAR STATION, UNIT NO. 1

SOUTH CAROLINA ELECTRIC & GAS COMPANY

SOUTH CAROLINA PUBLIC SERVICE AUTHORITY

DOCKET NO. 50-395

ENVIRONMENTAL PROTECTION PLAN

VIRGIL C. SUMMER NUCLEAR STATION

UNIT NO. 1

ENVIRONMENTAL PROTECTION PLAN

(NON-RADIOLOGICAL)

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1.0 Objectives of the Environmental Protection Plan

The Environmental Protection Plan (EPP) is to provide for protection of environmental values during construction and operation of the nuclear facility. The principal objectives of the EPP are as follows:

(1) Verify that the facility is operated in an environmentally acceptable manner, as established by the Final Environmental Statement (FES) and other NRC environmental impact assessments.

(2) Coordinate NRC requirements and maintain consistency with other Federal, State and local requirements for environmental protection.

(3) Keep NRC informed of the environmental effects of facility construction and operation and of actions taken to control those effects.

Environmental concerns identified in the FES which relate to water quality matters are regulated by means of the licensees' NPDES permit.

2.0 Environmental Protection Issues

In the FES-OL dated May 1981, the staff has considered the environmental impacts associated with the operation of the Virgil C. Summer Nuclear Station, Unit No. 1. No environmental issues were identified which require license conditions to resolve environmental concerns and to assure adequate protection of the environment.

Aquatic monitoring is addressed by the effluent limitations, monitoring requirements and demonstration studies contained in the effective NPDES permit issued by the South Carolina Department of Health and Environmental Control. The NRC will rely on that agency for regulations of matters involving water quality and aquatic biota.

3.0 Consistency Requirements

3.1 Facility Design and Operation

The licensees may make changes in facility design or operation or perform tests or experiments affecting the environment provided that such changes, tests or experiments do not involve an unreviewed environmental question. Changes in facility design or operation or performance of tests or experiments which do not affect the environment are not subject to this requirement. Activities governed by Section 3.3 are not subject to the requirements of this section.

Before engaging in unauthorized construction or operational activities which may affect the environment, the licensees shall prepare and record an environmental evaluation of such activity.[2] When the evaluation indicates that such activity involves an unreviewed environmental question, the licensees shall provide a written evaluation of such activities and obtain prior approval from the NRC.

A proposed change, test or experiment shall be deemed to involve an unreviewed environmental question if it concerns (1) a matter that may result in a significant increase in any adverse environmental impact previously evaluated in the Final Environmental Statement (FES) as modified by the staff's testimony to the Atomic Safety and Licensing Board, supplements to the FES, environmental impact appraisals, or in any decisions of the Atomic Safety and Licensing Board; or (2) a significant change in effluents or power level (in accordance with 10 CFR Part 51.5(b)(2)) or (3) a matter not previously reviewed and evaluated in the documents specified in (1) of this paragraph, which may have a significant adverse environmental impact.

The licensees shall maintain records of changes in facility design or operation and of tests and experiments carried out pursuant to this subsection. These records shall include a written

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[2]Activities are excluded from this requirement if all measurable nonradiological effects are confined to the on-site areas previously disturbed during site preparation and plant construction.

evaluation which provides bases for the determination that the change, test, or experiment does not involve an unreviewed environmental question.

3.2 Reporting Related to the NPDES Permits and State Certifications

Violations of the NPDES permit or State certification (pursuant to Section 401 of the Clean Water Act) shall be reported to the NRC by submittal of copies of the reports required by the NPDES permit or certification. South Carolina Electric & Gas Company shall also provide the NRC with a copy of the results of the following studies at the same time they are submitted to the permitting agency:

(1) Thermal Effects Study Plan

(2) Section 316(b) Demonstration Study

Changes and additions to the NPDES permit or the State certification shall be reported to the NRC within 30 days following the date the change is approved. If a permit or certification, in part or in its entirety, is appealed and stayed, the NRC shall be notified within 30 days following the date the stay is granted.

The NRC shall be notified of changes to the effective NPDES permit proposed by the licensees by providing NRC with a copy of the proposed change at the same time it is submitted to the permitting agency. South Carolina Electric & Gas Company shall provide the NRC with a copy of the application for renewal of the NPDES permit at the same time the application is submitted to the permitting agency.

3.3 Changes Required for Compliance with Other Environmental Regulations

Changes in facility design or operation and performance of tests or experiments which are required to achieve compliance with other Federal, State or local environmental regulations are not subject to the requirements of Section 3.1.

4.0 Environmental Conditions

4.1 Unusual or Important Environmental Events

Any occurrence of an unusual or important event that indicates or could result in significant environmental impact causally related to facility operation shall be recorded and promptly reported to the NRC within 24 hours followed by a written report within 30 days. No routine monitoring programs are required to implement this condition.

The written report shall (a) describe, analyze, and evaluate the event, including the extent and magnitude of the impact and facility operating characteristics, (b) describe the probable cause of the event, (c) indicate the action taken to preclude repetition of the event and to prevent similar occurrences involving similar components or systems, and (d) indicate the agencies notified and their preliminary responses.

Events reportable under this subsection which also require reports to other Federal, State or local agencies shall be reported in accordance with those reporting requirements in lieu of the requirements of this subsection. The NRC shall be provided with a copy of such reports at the same time they are submitted to the other agencies.

The following are examples of unusual or important events: excessive bird impaction events; onsite plant or animal disease outbreaks; mortality or unusual occurrence of any species protected by the Endangered Species Act of 1973; unusual fish kills; increase in nuisance organisms or conditions; and unanticipated or emergency discharge of waste water or chemical substance.

APPENDIX C

ADDITIONAL CONDITIONS
RENEWED OPERATING LICENSE NO. NPF-12

South Carolina Electric & Gas Company (the term licensee in Appendix C refers to South Carolina Electric & Gas Company) shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
137	The licensee is authorized to relocate certain Technical Specification requirements to licensee-controlled documents. Implementation of this amendment shall include the relocation of those technical specification requirements to the appropriate documents, as described in the licensee's application dated November 14, 1995, as supplemented by letters dated July 11, 1996, and July 24, 1997, and evaluated in the staff's Safety Evaluation attached to this amendment.	The amendment shall be implemented within 180 days from August 13, 1997