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1.0 DEFINITIONS

DEFINED TERMS

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

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3390 MWt.

OPERATIONAL MODE - MODE

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

ACTION

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

OPERABLE - OPERABILITY

SEE INSERT "A" ATTACHED

~~1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s), and when all necessary attendant instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).~~

REPORTABLE OCCURRENCE

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

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INSERT "A"

OPERABLE - OPERABILITY

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

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.4.1.

b. All equipment hatches are closed and sealed,

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

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

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

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

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

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

c. Digital computer channels - the exercising of the digital computer hardware using diagnostic programs and the injection of simulated process data into the channel to verify OPERABILITY.

CORE ALTERATION

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

SHUTDOWN MARGIN

SEE INSERT "A"

~~1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:~~

~~a. No change in part length control element assembly position, and~~

~~b. All full length control element assemblies (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.~~

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

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

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

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

INSERT "A"

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all control element assemblies are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

UNIDENTIFIED LEAKAGE

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

PRESSURE BOUNDARY LEAKAGE

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

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be the seal water flow ~~supplied to (or~~ from~~)~~ the reactor coolant pump seals.

AZIMUTHAL POWER TILT - T_q

1.18 AZIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

DOSE EQUIVALENT I-131

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

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

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

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

FREQUENCY NOTATION

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

REACTOR TRIP SYSTEM RESPONSE TIME

1.22 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

AXIAL SHAPE INDEX

1.24 The AXIAL SHAPE INDEX shall be the power generated in the lower half of the core less the power generated in the upper half of the core divided by the sum of these powers.

PHYSICS TESTS

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

SHIELD BUILDING INTEGRITY

1.27 SHIELD BUILDING INTEGRITY shall exist when:

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

b. The shield building filtration system is OPERABLE, and

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

SOFTWARE

1.~~28~~ 27 The digital computer SOFTWARE for the reactor protection system shall be the program codes including their associated data, documentation and procedures.

TABLE 1.1

OPERATIONAL MODES

OPERATIONAL MODE	REACTIVITY CONDITION, K_{eff}	% OF RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ ~~300~~ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ ~~300~~ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ ~~300~~ 350°F
4. HOT SHUTDOWN	< 0.99	0	~~300~~ 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.

** ~~Reactor vessel head unbolted or removed and fuel in the vessel.~~

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

TABLE 1.2

FREQUENCY NOTATION

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

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.~~30~~19.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.~~30~~19, be in HOT STANDBY within 1 hour.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate (adjusted for fuel rod dynamics) of the fuel shall be maintained less than or equal to 21.0 kw/ft.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate (adjusted for fuel rod dynamics) of the fuel has exceeded 21.0 kw/ft, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 2750 psia.

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

ACTION:

MODES 1 and 2

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within 1 hour.

MODES 3, 4 and 5

Whenever the Reactor Coolant System pressure has exceeded 2750 psia, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

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

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

ACTION:

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

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Linear Power Level - High		
a. Four Reactor Coolant Pumps Operating	≤ ~~(123)~~ (120)% of RATED THERMAL POWER	≤ ~~(123.712)~~ 121.3% of RATED THERMAL POWER
~~b. Three Reactor Coolant Pumps Operating~~	~~*~~	~~*~~
~~c. Two Reactor Coolant Pumps Operating - Same Loop~~	~~*~~	~~*~~
~~d. Two Reactor Coolant Pumps Operating - Opposite Loops~~	~~*~~	~~*~~
3. Logarithmic Power Level - High (1)	≤ ~~(0.75)~~ 0.89% of RATED THERMAL POWER	≤ ~~(0.819)~~ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ ~~(2345)~~ 2382 psia	≤ ~~(2353.887)~~ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ ~~(1740)~~ 1806 psia ~~(2)~~	≥ ~~(1686.75)~~ 1763 psia ~~(2)~~
6. Containment Pressure - High	≤ ~~(18.4)~~ 2.95 psia	≤ ~~(19.024)~~ 3.14 psia
7. Steam Generator Pressure - Low (3)	≥ ~~(728)~~ 729 psia ~~(3)~~	≥ ~~(706.6)~~ 711 psia ~~(3)~~
8. Steam Generator Level - Low (4)	≥ ~~(46.5)~~ 23% ~~(4)~~	≥ ~~(45.61)~~ 22.23% ~~(4)~~

~~*These values left blank pending NRC approval of operation with less than four reactor coolant pumps operating.~~

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TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High (5)	$\leq$ (~~20.3~~ 21) kw/ft ~~(5)~~	$\leq$ (~~20.3~~ 21) kw/ft ~~(5)~~
10. DNBR - Low (5)	$\geq$ (~~1.3~~ 1.19) ~~(5)~~	$\geq$ (~~1.3~~ 1.19) ~~(5)~~
11. Steam Generator Level - High (4)	$\leq$ ~~(93.6)~~ 90% ~~(4)~~	$\leq$ ~~(94.489)~~ 90.74% ~~(4)~~

TABLE NOTATION

(1) Trip may be manually bypassed above $\{10^{-4}\}$% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to $\{10^{-4}\}$% of RATED THERMAL POWER.

(2) Value may be decreased manually, to a minimum value of ~~{100}~~ 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer pressure and this value is maintained at less than or equal to ~~(200)~~ 400 psi; the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below {400} psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to ~~(500)~~ 400 psia.

(3) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to {200} psi; the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(4) % of the distance between steam generator upper and low level instrument nozzles.

(5) As stored within the Core Protection Calculator (CPC). Calculation of the trip setpoint includes measurement, calculational and processor uncertainties, and dynamic allowances. Trip may be manually bypassed below $\{10^{-4}\}$% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to $\{10^{-4}\}$% of RATED THERMAL POWER.

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

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NOTE

Summary statements

The ~~BASES~~ contained in this section provide the bases for the specifications of Section 2.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

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2.1.1 REACTOR CORE

The restrictions of these safety limits prevent overheating of the fuel cladding 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 (1) 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, and (2) maintaining the dynamically adjusted peak linear heat rate of the fuel at or less than 21 kw/ft which will not cause fuel centerline melting in any fuel rod.

First, by operating within the nucleate boiling regime of heat transfer, the heat transfer coefficient is large enough so that the maximum clad surface temperature is only slightly greater than the coolant saturation temperature. The upper boundary of the nucleate boiling regime is termed "departure from nucleate boiling" (DNB). At this point, there is a sharp reduction of the heat transfer coefficient, which would result in higher cladding temperatures and the possibility of cladding failure.

Correlations predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB ratio (DNBR), defined as the ratio of the predicted DNB heat flux at a particular core location to the actual heat flux at that location, is indicative of the margin to DNB. The minimum value of DNBR during normal operational occurrences is limited to ~~1.3~~ 1.19 for the ~~W-3~~ CE-1 correlation and is established as a Safety Limit.

Second, operation with a peak linear heat rate below that which would cause fuel centerline melting maintains fuel rod and cladding integrity. Above this peak linear heat rate level (i.e., with some melting in the center), fuel rod integrity would be maintained only if the design and operating conditions are appropriate throughout the life of the fuel rods. Volume changes which accompany the solid to liquid phase change are significant and require accomodation. Another consideration involves the redistribution of the fuel which depends on the extent of the melting and the physical state of the fuel rod at the time of melting. Because of the above factors, the steady state value of the peak linear heat rate which would not cause fuel centerline melting is established as a Safety Limit. To account for fuel rod dynamics (lags), the directly indicated linear heat rate is dynamically adjusted.

Limiting safety system settings for the Low DNBR, High Local Power Density, High Logarithmic Power Level, Low Pressurizer Pressure and High Linear Power Level trips, and limiting conditions for operation on DNBR and kw/ft margin are specified such that there is a high degree of confidence that the specified acceptable fuel design limits are not exceeded during normal operation and design basis anticipated operational occurrences.

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 Coolant System components are designed to Section III, 1971 ~~____~~ Edition, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the 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. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

The DNBR - Low and Local Power Density - High are digitally generated trip setpoints based on Limiting Safety System Settings of ~~1.30~~ 1.19 and ~~20.3~~ 21 kw/ft, respectively. Since these trips are digitally generated by the Core Protection Calculators, the trip values are not subject to drifts common to trips generated by analog type equipment. The Allowable Values for these trips are therefore the same as the Trip Setpoints.

To maintain the margins of safety assumed in the safety analyses, the calculations of the trip variables for the DNBR - Low and Local Power Density - High trips include the measurement, calculational and processor uncertainties and dynamic allowances as defined in ~~(reference applicable system descriptions and safety analyses).~~ (Later)

Manual Reactor Trip

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

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SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

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Linear Power Level-High

The Linear Power Level-High trip provides reactor core protection against rapid reactivity excursions which might occur as the result of an ejected CEA. This trip initiates a reactor trip at a linear power level of less than or equal to ~~(123.712)~~ 121.3% of RATED THERMAL POWER.

Logarithmic Power Level-High

The Logarithmic Power Level - High trip is provided to protect the integrity of fuel cladding and the Reactor Coolant System pressure boundary in the event of an unplanned criticality from a shutdown condition. A reactor trip is initiated by the Logarithmic Power Level - High trip at a THERMAL POWER level of less than or equal to ~~(0.819)~~ 0.96% of RATED THERMAL POWER unless this trip is manually bypassed by the operator. The operator may manually bypass this trip when the THERMAL POWER level is above 10^{-4}% of RATED THERMAL POWER; this bypass is automatically removed when the THERMAL POWER level decreases to 10^{-4}% of RATED THERMAL POWER.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to ~~(2353.887)~~ 2389 psia which is below the nominal lift setting (2500 psia) of the pressurizer safety valves and its operation avoids the undesirable operation of the pressurizer safety valves.

Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip is provided to trip the reactor and to assist the Engineered Safety Features System in the event of a Loss of Coolant Accident. During normal operation, this trip's setpoint is set at greater than or equal to ~~(1686.75)~~ 1763 psia. This trip's setpoint may be manually decreased, to a minimum value of ~~(100)~~ 300 psia, as pressurizer pressure is reduced during plant shutdowns, provided the margin between the pressurizer pressure and this trip's setpoint is maintained at less than or equal to ~~(200)~~ 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated concurrently with a safety injection. The setpoint for this trip is identical to the safety injection setpoint.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint is sufficiently below the full load operating point of approximately {900} psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This trip's setpoint may be manually decreased as steam generator pressure is reduced during plant shutdowns, provided the margin between the steam generator pressure and this trip's setpoint is maintained at less than or equal to {200} psi; this setpoint increases automatically as steam generator pressure increases until the trip setpoint is reached.

Steam Generator Level-Low

The Steam Generator Level-Low trip provides protection against a loss of feedwater flow incident and assures that the design pressure of the Reactor Coolant System will not be exceeded due to loss of the steam generator heat sink. This specified setpoint provides allowance that there will be sufficient water inventory in the steam generator at the time of the trip to provide a margin of at least 10 minutes before emergency feedwater is required.

Local Power Density-High

The Local Power Density-High trip is provided to prevent the linear heat rate (kw/ft) in the limiting fuel rod in the core from exceeding the fuel design limit in the event of any anticipated operational occurrence. The local power density is calculated in the reactor protective system utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore flux monitoring system;

b. Radial peaking factors from the position measurement for the CEAs;

c. Delta T power from reactor coolant temperatures and coolant flow measurements.

BASES

Local Power Density-High (Continued)

The local power density (LPD), the trip variable, calculated by the CPC incorporates uncertainties and dynamic compensation routines. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core peak LPD is sufficiently less than the fuel design limit such that the increase in actual core peak LPD after the trip will not result in a violation of the peak LPD Safety Limit. CPC uncertainties related to peak LPD are the same types used for DNBR calculation. Dynamic compensation for peak LPD is provided for the effects of core fuel centerline temperature delays (relative to changes in power density), sensor time delays, and protection system equipment time delays.

DNBR-Low

The DNBR - Low trip is provided to prevent the DNBR in the limiting coolant channel in the core from exceeding the fuel design limit in the event of anticipated operational occurrences. The DNBR - Low trip incorporates a low pressurizer pressure floor of ~~1750~~ 1825 psia. At this pressure a DNBR - Low trip will automatically occur. The DNBR is calculated in the CPC utilizing the following information:

a. Nuclear flux power and axial power distribution from the excore neutron flux monitoring system;

b. Reactor Coolant System pressure from pressurizer pressure measurement;

c. Differential temperature (Delta T) power from reactor coolant temperature and coolant flow measurements;

d. Radial peaking factors from the position measurement for the CEAs;

e. Reactor coolant mass flow rate from reactor coolant pump speed;

f. Core inlet temperature from reactor coolant cold leg temperature measurements.

The DNBR, the trip variable, calculated by the CPC incorporates various uncertainties and dynamic compensation routines to assure a trip is initiated prior to violation of fuel design limits. These uncertainties and dynamic compensation routines ensure that a reactor trip occurs when the actual core DNBR is sufficiently greater than ~~1.3~~ 1.19 such that the decrease in actual core

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

DNBR-Low (Continued)

DNBR after the trip will not result in a violation of the DNBR Safety Limit. CPC uncertainties related to DNBR cover CPC input measurement uncertainties, algorithm modelling uncertainties, and computer equipment processing uncertainties. Dynamic compensation is provided in the CPC calculations for the effects of coolant transport delays, core heat flux delays (relative to changes in core power), sensor time delays, and protection system equipment time delays.

The DNBR algorithm used in the CPC is valid only within the limits indicated below and operation outside of these limits will result in a CPC initiated trip.

a.	RCS Cold Leg Temperature-Low	≥ ~~(465)~~ 495 °F
b.	RCS Cold Leg Temperature-High	≤ ~~(605)~~ 580 °F
c.	Axial Shape Index-Positive	Not more positive than ~~+(0.6)~~ 0.5
d.	Axial Shape Index-Negative	Not more negative than ~~-(0.6)~~ 0.5
e.	Pressurizer Pressure-Low	≥ ~~(1750)~~ 1825 psia
f.	Pressurizer Pressure-High	≤ ~~(2400)~~ 2375 psia
g.	Integrated Radial Peaking Factor-Low	≥ ~~(~~1.28~~)~~
h.	Integrated Radial Peaking Factor-High	≤ ~~(~~4.28~~)~~
i.	Quality Margin-Low	≥ ~~(~~0~~)~~

Steam Generator Level-High

The Steam Generator Level-High trip is provided to protect the turbine from excessive moisture carry over. Since the turbine is automatically tripped when the reactor is tripped, this trip provides a reliable means for providing protection to the turbine from excessive moisture carry over. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting ~~is required to enhance~~ enhances the overall reliability of the Reactor Protection System.

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SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

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

[SEE INSERT "A", ATTACHED]

~~3.0.3 When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, the unit shall be placed in a MODE in which the specification does not apply by placing it, as applicable, in:~~

~~1. At least HOT STANDBY within 1 hour~~

~~2. At least HOT SHUTDOWN within the next 6 hours, and~~

~~3. At least COLD SHUTDOWN within the following 30 hours.~~

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

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

[SEE INSERT "B", ATTACHED]

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

INSERT "A"

3.0.3 In the event a Limiting Condition for Operation and/or associated ACTION requirements cannot be satisfied because of circumstances in excess of those addressed in the specification, the unit shall be placed in at least HOT STANDBY within 1 hour, in at least HOT SHUTDOWN within the next 6 hours, and in at least COLD SHUTDOWN within the following 30 hours unless corrective measures are completed that permit operation under the permissible ACTION statements for the specified time interval as measured from initial discovery or until the reactor is placed in a MODE in which the specification is not applicable. Exceptions to these requirements shall be stated in the individual specifications.

INSERT "B"

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

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APPLICABILITY

SURVEILLANCE REQUIREMENTS

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

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

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

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

DO NOT APPLY TO

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

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

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

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

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

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APPLICABILITY

SURVEILLANCE REQUIREMENTS (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
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.

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3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

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

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to ~~(5.0)~~% 5.15 delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than ~~(5.0)~~ 5.15% delta k/k, immediately initiate and continue boration at greater than or equal to {40} gpm of a solution containing ~~(1731)~~ 1720 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 ~~(5.0)~~% 5.15 delta k/k:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA 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 CEA(s).

b. When in MODE~~S~~ 1 or ~~[illegible]~~ MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Transient Insertion 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 CEA position is within the limits of Specification {3.1.3.6}.

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the Transient Insertion Limits of Specification 3.1.3.6.

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

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

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be greater than or equal to 2.0% ~~(1.0)%~~ delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 2.0% ~~(1.0)%~~ delta k/k, immediately initiate and continue boration at greater than or equal to {40} gpm of a solution containing {1731} ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be greater than or equal to ~~(1.0)%~~ 2.0 delta k/k:

~~a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).~~

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

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

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than {0.5} x 10^{-4} delta k/k/°F, and

b. Less negative than ~~{-3.5}~~ -3.3 x 10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

4.1.1.3.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

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

b. At any THERMAL POWER, within 7 EFPD after ~~reaching a~~ RATED THERMAL POWER equilibrium boron concentration ~~of (800)~~ decreases to 900 ppm.

c. At any THERMAL POWER, within 7 EFPD after ~~reaching a~~ RATED THERMAL POWER equilibrium boron concentration ~~of~~ decreases to {300} ppm.

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

#See Special Test Exception 3.10.2.

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REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to ~~(525)~~ °F when the reactor is critical.

(LATER)

APPLICABILITY: MODES 1 and 2#.

ACTION:

520 With a Reactor Coolant System operating loop temperature (T_{avg}) less than ~~(525)~~ °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 ~~(525)~~ °F:

(LATER)

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 {535}°F.

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

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE:

SEE INSERT "A" ATTACHED

~~a. A flow path from the boric acid makeup tank via either a boric acid makeup pump or a gravity feed connection and charging pump to the Reactor Coolant System if only the boric acid makeup tank in Specification (3.1.2.7.a) is OPERABLE, or~~

b. The flow path from the refueling water tank via either a charging pump or a high pressure safety injection pump to the Reactor Coolant System if only the refueling water tank in Specification 3.1.2.7.b is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path is above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup tanks is used.

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

Insert 'A'

a. A flow path from either boric acid makeup tank via either one of the boric acid makeup pumps, the blending tee or the gravity feed connection, and any charging pump to the RCS if only a boric acid makeup tank in paragraph 3.1.2.7a is OPERABLE, or

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 At least two of the following three boron injection flow paths ~~and one associated heat tracing circuit~~ shall be OPERABLE:

a. Two flow paths from the boric acid makeup tanks via either a boric acid makeup pump or a gravity feed connection, and a charging pump to the Reactor Coolant System, and

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid makeup tanks is above the temperature limit line shown on Figure 3.1-1.

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

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

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

~~4.1.2.3 At least the above required pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the charging pump (if applicable) develops a discharge pressure of greater than or equal to ____ psig or the high pressure safety injection pump (if applicable) develops a discharge pressure of greater than or equal to ____ psig when tested pursuant to Specification 4.0.5.~~

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

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REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

~~4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE by verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to ____ psig when tested pursuant to Specification 4.0.5.~~

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

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BORIC ACID MAKEUP PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid makeup pump shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if only the flow path through the boric acid pump in Specification 3.1.2.1.a is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid makeup pump OPERABLE as required to complete the flow path of Specification 3.1.2.1.a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

~~4.1.2.5 The above required boric acid makeup pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to ____ psig when tested pursuant to Specification 4.0.5.~~

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

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BORIC ACID MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2.a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2.a inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours ~~and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F~~; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

~~4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump(s) develop a discharge pressure of greater than or equal to ___ psig when tested pursuant to Specification 4.0.5.~~

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

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank ~~and at least one associated heat tracing circuit~~ with the tank contents in accordance with Figure [3.1-1].

b. The refueling water tank with:

1. A minimum contained borated water volume of 5465 ~~(35,250)~~ gallons,

2. A minimum boron concentration of 1720 ~~(1731)~~ ppm, and

3. A minimum solution temperature of {35}°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.7 The above required borated water source shall be demonstrated OPERABLE~~:~~ at least once per 7 days by:

~~a. At least once per 7 days by:~~

a. ~~1.~~ Verifying the boron concentration of the water,

b. ~~2.~~ Verifying the contained borated water volume of the tank, and

c. Verifying solution TEMPERATURE

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

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

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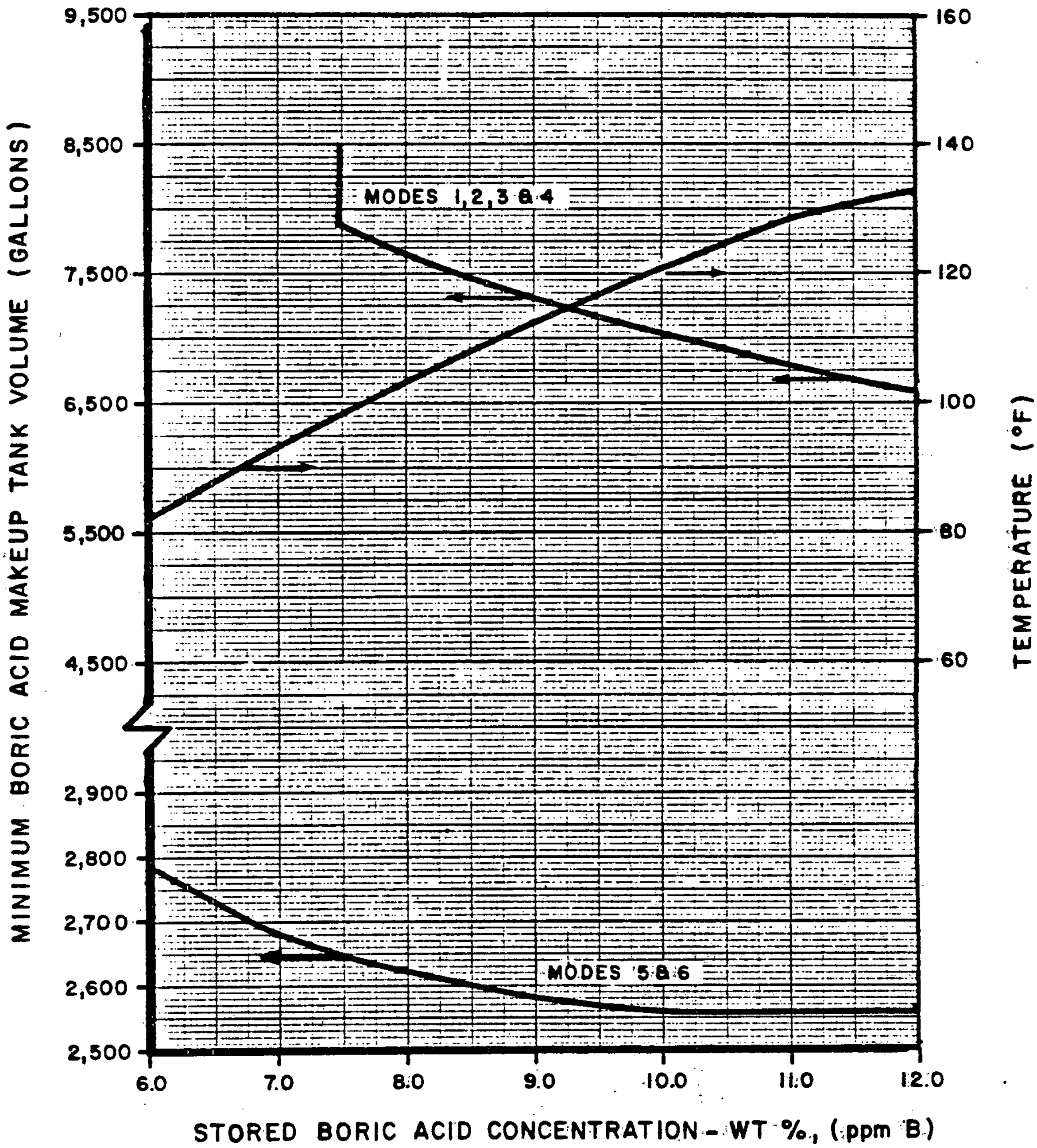

Figure 3.1-1

MINIMUM BORIC ACID STORAGE TANK VOLUME AND TEMPERATURE AS A FUNCTION OF STORED BORIC ACID CONCENTRATION

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 Each of the following borated water sources shall be OPERABLE:

a. At least one boric acid makeup tank ~~and at least one associated heat tracing circuit~~ per tank with the contents of the tank in accordance with Figure {3.1-1}, and

b. The refueling water STORAGE tank with:

1. A ~~contained~~ (LATER) borated water volume of between 355,000 ~~(464,900)~~ and ~~(500,500)~~ gallons,

2. Between 1720 ~~(1731)~~ and 2300 ~~(2250)~~ ppm of boron, and

3. A minimum solution temperature of {35}°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours ~~and borated to a SHUTDOWN MARGIN equivalent to at least 1% delta k/k at 200°F;~~ restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.8 Each borated water sources shall be demonstrated OPERABLE~~:~~ AT LEAST ONCE PER 7 DAYS by:

~~a. At least once per 7 days by:~~

a. ~~1.~~ Verifying the boron concentration in the water,

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

c. ~~3.~~ Verifying the boric acid makeup tank solution temperature.

~~b. At least once per 24 hours by verifying the RWT temperature when the (outside) air temperature is less than (35)°F.~~

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and regulating) CEAs, and all part length CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With one or more full length CEAs 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 at least HOT STANDBY within 6 hours.

b. With one full length CEA inoperable due to causes other than addressed by ACTION a., above, ~~and inserted beyond the Long Term Steady State Insertion Limits~~ but within its above specified alignment requirements, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6.

c. With one full length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements ~~and either fully withdrawn or within the Long Term Steady State Insertion Limits if in full length CEA group 6,~~ operation in MODES 1 and 2 may continue, if: (SEE INSERT "A" NEXT PAGE)

d. With one or more full length or part length CEAs misaligned from any other CEAs in its group by more than 7 inches but less than or equal to 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

*See Special Test Exceptions 3.10.2 and 3.10.4.

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INSERT "A"

1. The inoperable CEA is in any shutdown or regulating group except group __ (LATER) and is fully withdrawn, or
2. The inoperable CEA is in group __ (LATER) and is within the Long Term Steady State Insertion Limits.

ACTION: (Continued)

a) Within one hour the remainder of the CEA(s) in the group(s) with the inoperable CEA(s) shall be aligned to within 7 inches of the inoperable CEA(s) while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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

Otherwise, be in at least HOT STANDBY within 6 hours.

e. With one full length or part length CEA misaligned from any other CEA in its group by more than 19 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA is either:

1. Restored to OPERABLE status within its above specified alignment requirements, or

2. Declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue pursuant to the requirements of Specification 3.1.3.6 provided:

a) Within one hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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

Otherwise, be in at least HOT STANDBY within 6 hours.

f. With one part length CEA inoperable and inserted in the core, operation may continue provided the alignment of the inoperable part length CEA is maintained within 7 inches (indicated position) of all other part lengtn CEAs in its group.

g. With more than one full length or part length CEA inoperable or misaligned from any other CEA in its group by more than 19 inches (indicated position), be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length and part length CEA shall be determined to be within 7 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when one CEAC is inoperable or when both CEACs are inoperable, then verify the individual CEA positions at least once per 4 hours.

4.1.3.1.2 Each full length CEA not fully inserted and each part length CEA which is inserted in the core shall be determined to be OPERABLE by movement of at least 5 inches in any one direction at least once per 31 days.

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 At least two of the following three CEA position indicator channels shall be OPERABLE for each CEA:

a. CEA Reed Switch Position Transmitter (RSPT 1) with the capability of determining the absolute CEA positions within 5 inches,

b. CEA Reed Switch Position Transmitter (RSPT 2) with the capability of determining the absolute CEA positions within 5 inches, and

c. The CEA pulse counting position indicator channel.

APPLICABILITY: MODES 1 and 2.

ACTION:

With a maximum of one CEA per CEA group having only one of the above required CEA position indicator channels OPERABLE, within 6 hours either:

a. Restore the inoperable position indicator channel to OPERABLE status, or

b. Be in at least HOT STANDBY, or

c. Position the CEA group(s) with the inoperable position indicator(s) at its fully withdrawn position while maintaining the requirements of Specifications 3.1.3.1 and 3.1.3.6. Operation may then continue provided the CEA group(s) with the inoperable position indicator(s) is maintained fully withdrawn, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2, and each CEA in the group(s) is verified fully withdrawn at least once per 12 hours thereafter by its "Full Out" limit.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each of the above required position indicator channels shall be determined to be OPERABLE by verifying that for the same CEA, the position indicator channels agree within 5 inches of each other at least once per 12 hours.

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA Reed Switch Position Transmitter indicator channel shall be OPERABLE for each shutdown, regulating or part length CEA not fully inserted.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

* With the reactor trip breakers in the closed position.

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CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be less than or equal to {3.0} seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position~~, with:~~

~~a. T_{avg} greater than or equal to (525)°F, and~~

~~b. All reactor coolant pumps operating.~~

APPLICABILITY: MODES 1 and 2.

ACTION:

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

~~b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.~~

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality (WITH $T_{avg} \geq 520$°F AND ALL REACTOR COOLANT PUMPS OPERATING):

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

b. For specifically affected individuals CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to the Full Out position.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than the Full Out position, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Withdraw the CEA to the Full Out position, or

b. Declare the CEA inoperable and apply Specification 23.1.3.12.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to the Full Out position:

a. Within 15 minutes prior to withdrawal of any CEAs in regulating groups during an approach to reactor criticality, and

b. At least once per 12 hours thereafter.

---

*See Special Test Exception 3.10.2.

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

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REACTIVITY CONTROL SYSTEMS

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on Figure 3.1-2, with CEA insertion between the Long Term Steady State Insertion Limits and the Transient Insertion Limits restricted to:

a. Less than or equal to 4 hours per 24 hour interval, and,

~~b. Less than or equal to 5 Effective Full Power Days per 30 Effective Full Power Day interval, and~~

b. ~~c.~~ Less than or equal to 14 Effective Full Power Days per calendar year.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Transient Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within two hours either:

1. Restore the regulating CEA groups to within the limits, or

2. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the CEA group position using the above figure.

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals greater than 4 hours per 24 hour interval, operation may proceed provided either:

1. The Short Term Steady State Insertion Limits of Figure 3.1-2 are not exceeded, or

2. Any subsequent increase in THERMAL POWER is restricted to less than or equal to 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.4.

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

ACTION: (Continued)

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Transient Insertion Limits for intervals ~~greater than 5 EFPD per 30 EFPD interval or~~ greater than 14 EFPD per calendar year, either:

1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within two hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Transient Insertion Limits at least once per 12 hours except during time intervals when the PDIL Auctioneer Alarm Circuit is inoperable, then verify the individual CEA positions at least once per 4 hours. The accumulated times during which the regulating CEA groups are inserted beyond the Long Term Steady State Insertion Limits but within the Transient Insertion Limits shall be determined at least once per 24 hours.

FIGURE 3.1-2
(LATER)

REACTIVITY CONTROL SYSTEMS

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the part length CEA group shall be restricted to prevent the neutron absorber section of the part length CEA group from covering any axial segment of the fuel assemblies for a period in excess of 7 out of any 30 EFPD period.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the neutron absorber section of the part length CEA group covering any axial segment of the fuel assemblies for a period exceeding 7 out of any 30 EFPD period, either:

a. Reposition the part length CEA group to satisfy the above limit within 2 hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.7 The position of the part length CEA group shall be determined at least once per 12 hours.

DELETE

REACTIVITY CONTROL SYSTEMS

PART LENGTH CEA INSERTION LIMITS (ALTERNATE if required by DNB considerations)

LIMITING CONDITION FOR OPERATION

3.1.3.8 All part length CEAs shall be withdrawn to at least (176) steps.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With a maximum of one PLCEA withdrawn to less than (176) steps, either:

a. Withdraw the PLCEA to at least (176) steps within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.8 Each part length CEA shall be determined withdrawn to at least (176) steps by:

a. Verifying the positions of the PLCEAs prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER, and

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

*See Special Test Exception 3.10.2.

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

(LATER)

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1 above (20)% of RATED THERMAL POWER.

ACTION:

With the linear heat rate exceeding its limits, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on kw/ft; or (2) when the COLSS is not being used, any OPERABLE Local Power Density channel exceeding the linear heat rate limit, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

(LATER)

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits when THERMAL POWER is above (20)% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the linear heat rate, as indicated on all OPERABLE Local Power Density channels, is within the limit shown on Figure 3.2-1.

4.2.1.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on kw/ft.

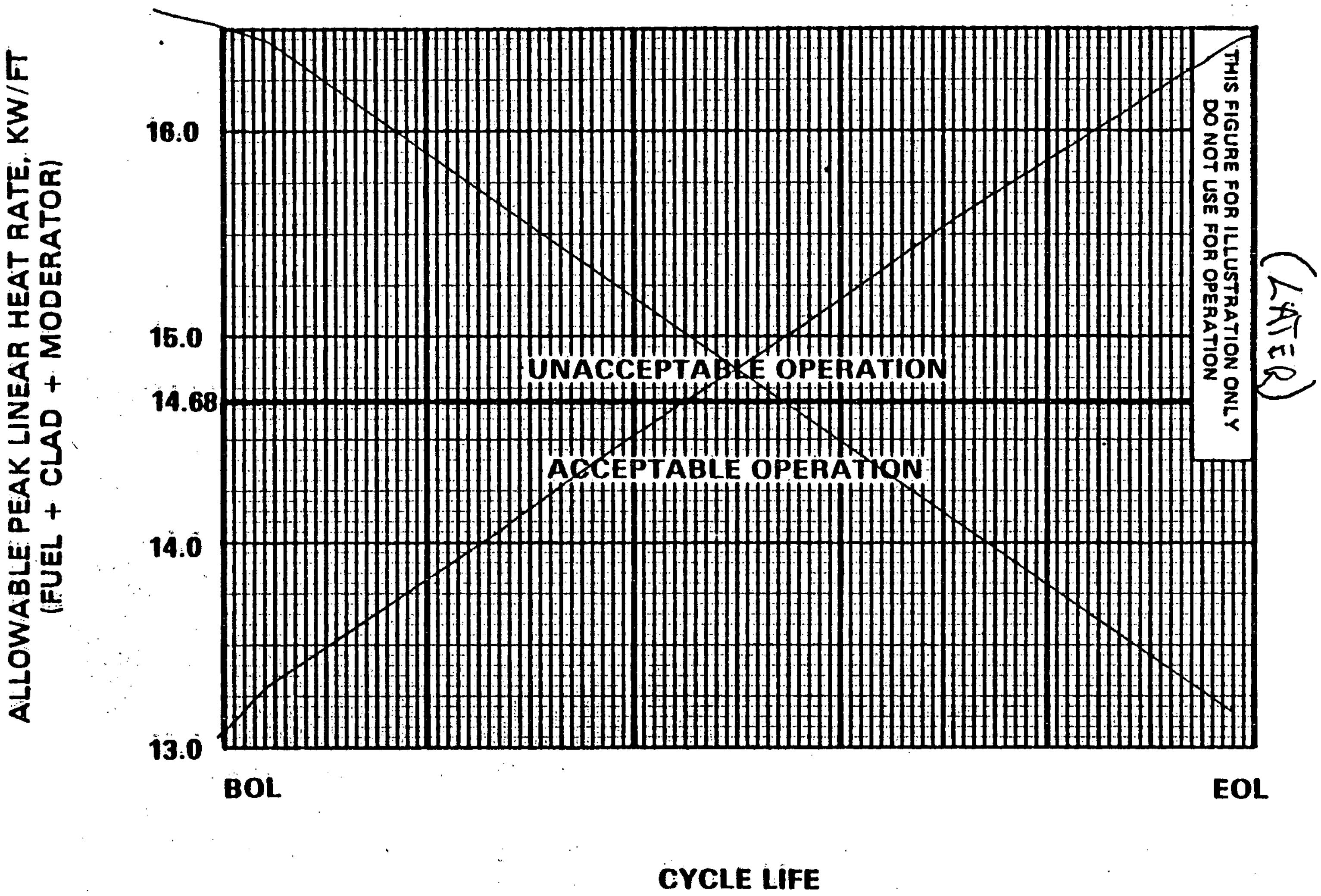

FIGURE 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

POWER DISTRIBUTION LIMITS

3/4.2.2 RADIAL PEAKING FACTORS

LIMITING CONDITION FOR OPERATION

(LATER)

3.2.2 The measured planar radial peaking factors (F_r^m) shall be less than or equal to the planar radial peaking factors (F_r^c) used in the Core Operating Limit Supervisory System (COLSS) and in the Core Protection Calculators (CPC).

APPLICABILITY: MODE 1 above 20% of RATED THERMAL POWER.*

ACTION:

With a F_r^m exceeding a corresponding F_r^c, within 6 hours either:

a. Adjust the CPC addressable constants to increase the multiplier applied to planar radial peaking by a factor equivalent to greater than or equal to F_r^m/F_r^c restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F_r^m/F_r^c) - 1.0] \times 100\%$ is maintained; or

b. Adjust the affected planar radial peaking factors (F_r^c) used in the COLSS and CPC to a value greater than or equal to the measured planar radial peaking factors (F_r^m) or

c. Be in at least HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

(LATER)

4.2.2.2 The measured planar radial peaking factors (F_r^m) obtained by using the incore detection system, shall be determined to be less than or equal to the planar radial peaking factors (F_r^c), used in the COLSS and CPC at the following intervals:

a. After each fuel loading with THERMAL POWER greater than (40)% but prior to operation above (70)% of RATED THERMAL POWER, and

b. At least once per 31 days of accumulated operation in MODE 1.

*See Special Test Exception 3.10.2.

3/4.2.3 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

(LATER)

~~3.2.3 The AZIMUTHAL POWER TILT (T_q) shall be less than or equal to the AZIMUTHAL POWER TILT Allowance used in the Core Protection Calculators (CPCs).~~

~~APPLICABILITY: MODE 1 above (20)% of RATED THERMAL POWER.*~~

~~ACTION:~~

~~a. With the measured AZIMUTHAL POWER TILT determined to exceed the AZIMUTHAL POWER TILT Allowance used in the CPCs but less than or equal to 0.10, within two hours either correct the power tilt or adjust the AZIMUTHAL POWER TILT Allowance used in the CPCs to greater than or equal to the measured value.~~

~~b. With the measured AZIMUTHAL POWER TILT determined to exceed 0.10:~~

~~1. Due to misalignment of either a part length or full length CEA, within 30 minutes verify that the Core Operating Limit Supervisory System (COLSS) (when COLSS is being used to monitor the core power distribution per Specifications 4.2.1 and 4.2.4) is detecting the CEA misalignment.~~

~~2. Verify that the AZIMUTHAL POWER TILT 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 Linear Power Level - 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 AZIMUTHAL POWER TILT 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.~~

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SURVEILLANCE REQUIREMENTS

(LATER)

4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above (20)% of RATED THERMAL POWER by:

a. Continuously monitoring the tilt with COLSS when the COLSS is OPERABLE.

b. Calculating the tilt at least once per 12 hours when the COLSS is inoperable.

c. Verifying at least once per 31 days, that the COLSS Azimuthal Tilt Alarm is actuated at an AZIMUTHAL POWER TILT greater than the AZIMUTHAL POWER TILT Allowance used in the CPCs.

d. Using the incore detectors at least once per 31 days to independently confirm the validity of the COLSS calculated AZIMUTHAL POWER TILT.

3/4.2.4 DNBR MARGIN

LIMITING CONDITION FOR OPERATION

(LATER)

3.2.4 The DNBR margin shall be maintained by operating within the region of acceptable operation of Figure 3.2-2 or 3.2-3, as applicable.

APPLICABILITY: MODE 1 above (20)% of RATED THERMAL POWER.

ACTION:

With operation outside of the region of acceptable operation, as indicated by either (1) the COLSS calculated core power exceeding the COLSS calculated core power operating limit based on DNBR; or (2) when the COLSS is not being used, any OPERABLE Low DNBR channel exceeding the DNBR limit, within 15 minutes initiate corrective action to reduce the DNBR to within the limits and either:

a. Restore the DNBR to within its limits within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The DNBR shall be determined to be within its limits when THERMAL POWER is above (20)% of RATED THERMAL POWER by continuously monitoring the core power distribution with the Core Operating Limit Supervisory System (COLSS) or, with the COLSS out of service, by verifying at least once per 2 hours that the DNBR, as indicated on all OPERABLE DNBR channels, is within the limit shown on Figure 3.2-3.

4.2.4.3 At least once per 31 days, the COLSS Margin Alarm shall be verified to actuate at a THERMAL POWER level less than or equal to the core power operating limit based on DNBR.

SURVEILLANCE REQUIREMENTS (Continued)

(LATER)

4.2.4.4 The following DNBR penalty factors shall be verified to be included in the COLSS and CPC DNBR calculations at least once per 31 days:

Burnup $(\frac{GWD}{MTU})$	DNBR Penalty (%)
0-3.1	(0.0)
3.1-5	(2.0)
5-10	(5.9)
10-15	(8.8)
15-20	(11.4)
20-25	(13.6)
25-30	(15.6)
30-35	(17.4)

(LATER)

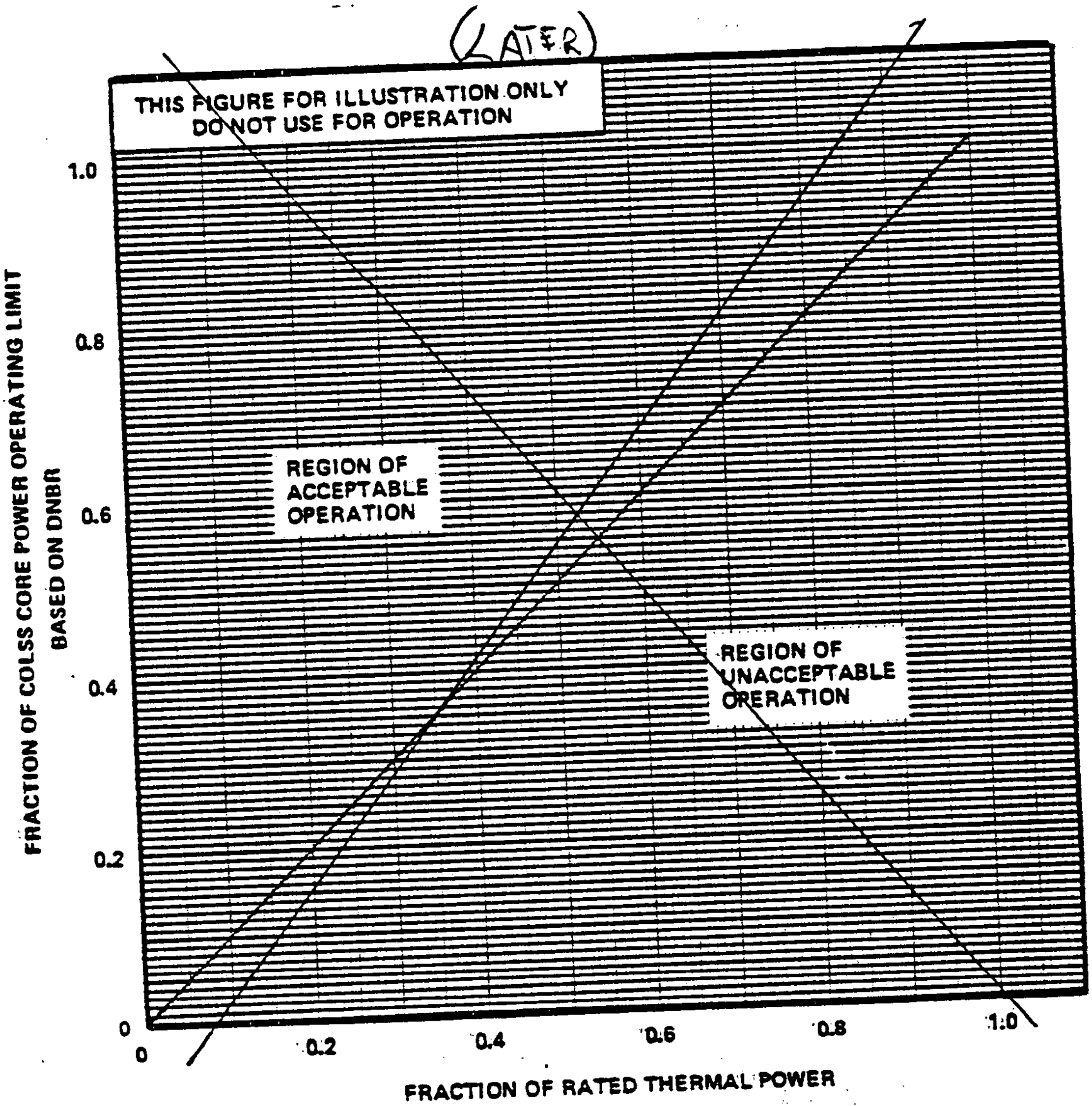

Figure 3.2-2
DNBR Margin Operating Limit Based on COLSS

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(LATER)

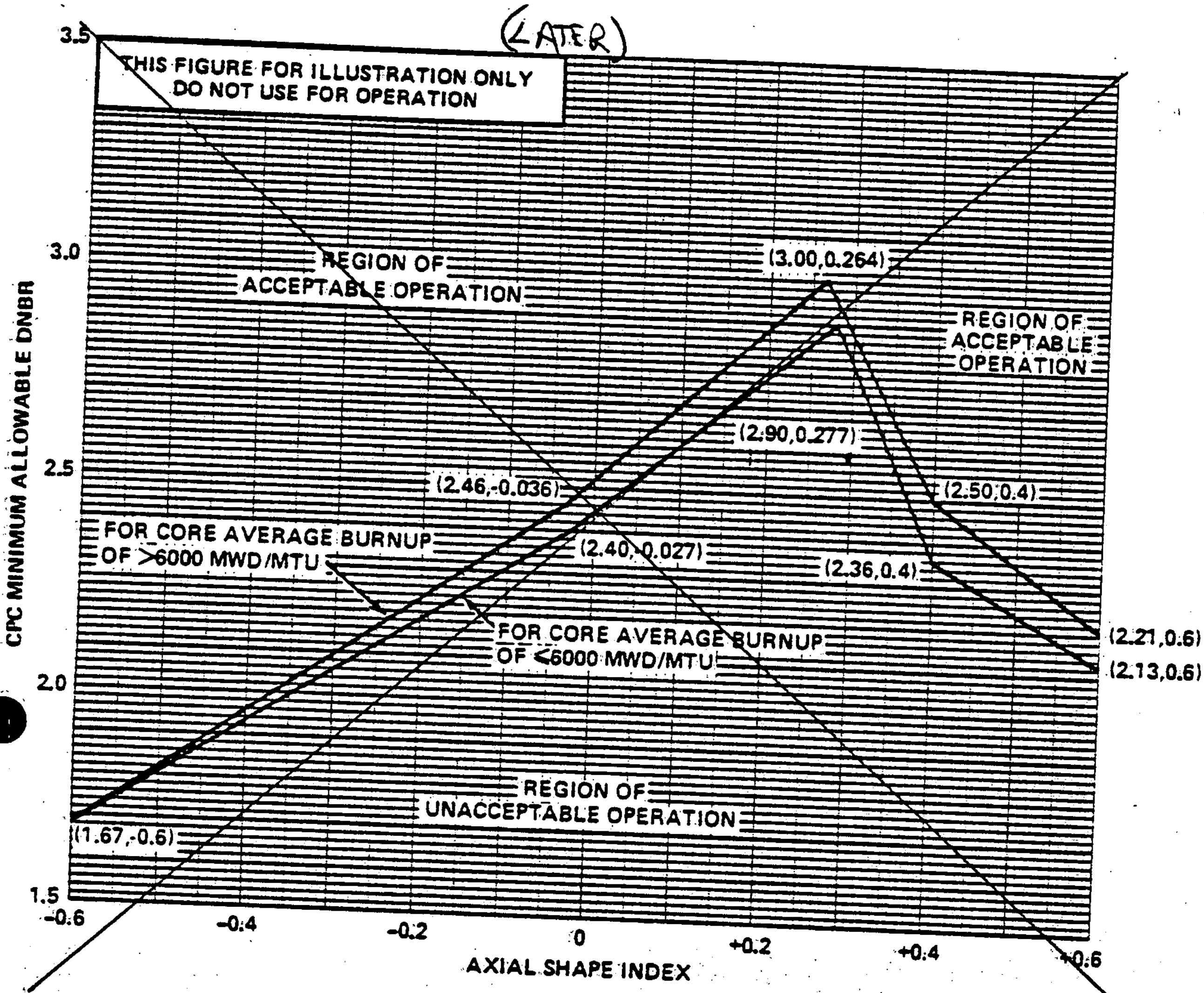

Figure 3.2-3
DNBR Margin Operating Limit Based on Core Protection Calculators
(COLSS Out of Service)

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION (LATER)

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to (120.4×10^6) lbm/hr.

APPLICABILITY: MODE 1.

ACTION:

With the actual Reactor Coolant System total flow rate determined to be less than the above limit, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5 The actual Reactor Coolant System total flow rate shall be determined to be within its limit at least once per 12 hours.

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3/4.2.6 CORE AVERAGE COOLANT TEMPERATURE

LIMITING CONDITION FOR OPERATION (LATER)

3.2.6 The core average coolant temperature (T_{avg}) shall be less than or equal to (588.2)°F.

APPLICABILITY: MODE 1

ACTION:

With the core average coolant temperature exceeding its limit, restore the temperature 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

4.2.6 The core average coolant temperature shall be determined to be within its limit at least once per 12 hours.

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3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protective instrumentation channels and bypasses 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 protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

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

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

4.3.1.4 The isolation characteristics of each CEA isolation amplifier and each optical isolator for CEA Calculator to Core Protection Calculator data transfer shall be verified at least once per 18 months during the shutdown per the following tests:

a. For the CEA position isolation amplifiers:

1. With 120 volts AC {60 Hz} applied for at least 30 seconds across the output, the reading on the input does not exceed {0.015} volts DC.

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SURVEILLANCE REQUIREMENTS (Continued)

2. With 120 volts AC (60 Hz) applied for at least 30 seconds across the input, the reading on the output does not exceed (8) volts DC.

b. For the optical isolators: Verify that the input to output insulation resistance is greater than (10) megohms when tested using a megohmmeter on the 500 volt DC range.

4.3.1.5 The Core Protection Calculator System shall be determined OPERABLE at least once per 12 hours by verifying that less than three auto restarts have occurred on each calculator during the past 12 hours.

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~~CE-STS~~ SAN ONOFRE-UNIT 2

TABLE 3.3-1

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2 and *	1
2. Linear Power Level - High	4	2	3	1, 2	2#
3. Logarithmic Power Level-High					
a. Startup and Operating	4	2(a)(d)	3	1, 2, and *	2#
b. Shutdown	4	0	2	3, 4, 5	3
4. Pressurizer Pressure - High	4	2	3	1, 2	2#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#
6. Containment Pressure - High	4	2	3	1, 2	2#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#
12. Reactor Protection System Logic	2	1	2	1, 2, and *	~~4~~ 2
13. Reactor Trip Breakers	2	1	2	1, 2, and *	~~4~~ 2
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2# and ~~6~~ 5
15. CEA Calculators	2	1	2(e)	1, 2	~~5~~ 4 and ~~6~~ 5

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TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position, the CEA drive system capable of CEA withdrawal, and fuel in the reactor vessel.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed above 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is less than or equal to 10^{-4}% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer pressure is greater than or equal to 500 psia.

(c) Trip may be manually bypassed below 10^{-4}% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 10^{-4}% of RATED THERMAL POWER. During testing pursuant to Special Test Exception 3.10.3, trip may be manually bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is greater than or equal to 1% of RATED THERMAL POWER.

(d) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(e) See Special Test Exception 3.10.2.

(f) Each channel shall be comprised of two trip breakers; actual trip logic shall be one-out-of-two taken twice.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

[SEE INSERT "A", ATTACHED]

~~ACTION 2 - With the number of channels OPERABLE 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.~~

TABLE 3.3-1 (Continued) [SEE INSERT "A" ATTACHED]

ACTION STATEMENTS

b. All functional units receiving an input from the tripped channel are also placed in the tripped condition within 1 hour.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.1.1.

ACTION 5 - a. With one CEAC inoperable, operation may continue for up to 7 days provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group.

b. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to greater than or equal to 8% of RATED THERMAL POWER.

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "Off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 1. a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 6 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

TABLE 3.3-1 (Continued)

TABLE NOTATION

ACTION 2 - A. With the number of OPERABLE channels one less than the Total Number of Channels, the inoperable channel shall be placed in the bypass condition.

B. With the number of channels OPERABLE one less than the minimum channels OPERABLE:

1. Place one inoperable channel in the bypassed condition and one inoperable channel in the tripped condition within 1 hour.

2. Operation may continue with one inoperable channel in the bypassed condition and one inoperable channel in the tripped condition provided all functional units receiving an input from the bypassed channel are also placed in the bypassed condition within 1 hour and all functional units receiving an input from the tripped inoperable channel are tripped.

C. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass all associated functional units as listed below:

	Process Measurement Circuit	Functional Unit Bypassed
1.	Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low
2.	Pressurizer Pressure - High	Pressurizer Pressure - High Local Power Density - High DNBR - Low
3.	Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS 1 and 2)
5.	Steam Generator Level	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)
6.	Core Protection Calculator	Local Power Density - High DNBR - Low

INSERT A

TABLE NOTATION

D. With an inoperable channel inthe bypass condition, the channel shall be returned to the operable condition as soon as practical, but not longer than 90 days for failures outside containment or not longer than 18 months for failures inside containment, and the bypass removed.

ACTION 3 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter. Within 48 hours return 1 channel to operation or provide an alternate source level monitoring system.

ACTION 4 - A. With one CEAC inoperable, operation may continue ~~for up to 7 days~~ provided that at least once per 4 hours, each CEA is verified to be within 7 inches (indicated position) of all other CEAs in its group.

B. With both CEACs inoperable, operation may continue provided that:

1. Within 1 hour the margins required by Specifications 3.2.1 and 3.2.4 are increased and maintained at a value equivalent to greater than or equal to 8% of RATED THERMAL POWER.

2. Within 4 hours:

a) All full length and part length CEA groups are withdrawn to and subsequently maintained at the "Full Out" position, except during surveillance testing pursuant to the requirements of Specification 4.1.3.1.2 or for control when CEA group 6 may be inserted no further than 127.5 inches withdrawn.

b) The "RSPT/CEAC Inoperable" addressable constant in the CPCs is set to the inoperable status.

c) The Control Element Drive Mechanism Control System (CEDMCS) is placed in and subsequently maintained in the "off" mode except during CEA group 6 motion permitted by a) above, when the CEDMCS may be operated in either the "Manual Group" or "Manual Individual" mode.

3. At least once per 4 hours, all full length and part length CEAs are verified fully withdrawn except during surveillance testing pursuant to Specification 4.1.3.1.2 or during surveillance testing pursuant to Specification 4.1.3.1.2 or during insertion of CEA group 6 as permitted by 1. a) above, then verify at least once per 4 hours that the inserted CEAs are aligned within 7 inches (indicated position) of all other CEAs in its group.

ACTION 5 - With three or more auto restarts of one non-bypassed calculator during a 12-hour interval, demonstrate calculator OPERABILITY by performing a CHANNEL FUNCTIONAL TEST within the next 24 hours.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Linear Power Level - High	$\leq$ 0.40 seconds*
3. Logarithmic Power Level - High	$\leq$ 0.45 seconds*
4. Pressurizer Pressure - High	$\leq$ 1.50 seconds
5. Pressurizer Pressure - Low	$\leq$ 0.90 seconds
6. Containment Pressure - High	$\leq$ 0.90 seconds
7. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
8. Steam Generator Level - Low	$\leq$ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ (LATER) seconds*
b. CEA Positions	$\leq$ (LATER) seconds**
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	$\leq$ (LATER) seconds*
b. CEA Positions	$\leq$ (LATER) seconds**
c. Cold Leg Temperature	$\leq$ (LATER) seconds##
d. Hot Leg Temperature	$\leq$ (LATER) seconds##
e. Primary Coolant Pump Shaft Speed	$\leq$ (LATER) seconds#
f. Reactor Coolant Pressure from Pressurizer	$\leq$ (LATER) seconds

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
11. Steam Generator Level - High	Not Applicable
12. Reactor Protection System Logic	Not Applicable
13. Reactor Trip Breakers	Not Applicable
14. Core Protection Calculators	Not Applicable
15. CEA Calculators	Not Applicable

*Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel. ~~(This provision is not applicable to CP's docketed after January 1, 1978. See Regulatory Guide 1.118, November 1977.)~~

**Response time shall be measured from the onset of a single CEA drop.

#Response time shall be measured from the onset of a 2 out of 4 Reactor Coolant Pump coastdown.

##Based on a resistance temperature detector (RTD) response time of less than or equal to 6.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Linear Power Level - High	S	D(2,4),M(3,4), Q(4)	M	1, 2
3. Logarithmic Power Level - High	S	R(4)	M and S/U(1)	1, 2, 3, 4, 5, and *
4. Pressurizer Pressure - High	S	R	M	1, 2
5. Pressurizer Pressure - Low	S	R	M	1, 2
6. Containment Pressure - High	S	R	M	1, 2
7. Steam Generator Pressure - Low	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	D(2,4), R(4,5)	M, R(6)	1, 2
10. DNBR - Low	S	S(7), D(2,4), M(8), R(4,5)	M, R(6)	1, 2
11. Steam Generator Level - High	S	R	M	1, 2
12. Reactor Protection System Logic	N.A.	N.A.	M	1, 2, and *

TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2, and *
14. Core Protection Calculators	S	D(2,4),R(4,5)	M,R(6)	1, 2
15. CEA Calculators	S	R	M,R(6)	1, 2

TABLE 4.3-1 (Continued)

TABLE NOTATION

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

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

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust the Linear Power Level signals and the CPC addressable constant multipliers to make the CPC delta T power and CPC nuclear power calculations agree with the calorimetric calculation if absolute difference is greater than 2%. During PHYSICS TESTS, these daily calibrations may be suspended provided these calibrations are performed upon reaching each major test power plateau and prior to proceeding to the next major test power plateau.

(3) - Above 15% of RATED THERMAL POWER, verify that the linear power subchannel gains of the excore detectors are consistent with the values used to establish the shape annealing matrix elements in the Core Protection Calculators.

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

(5) - After each fuel loading and prior to exceeding 70% of RATED THERMAL POWER, the incore detectors shall be used to determine the shape annealing matrix elements and the Core Protection Calculators shall use these elements.

(6) - This CHANNEL FUNCTIONAL TEST shall include the injection of simulated process signals into the channel as close to the sensors as practicable to verify OPERABILITY including alarm and/or trip functions.

(7) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by either using the reactor coolant pump differential pressure instrumentation (conservatively compensated for measurement uncertainties) or by calorimetric calculations (conservatively compensated for measurement uncertainties) and if necessary, adjust the CPC addressable constant flow coefficients such that each CPC indicated flow is less than or equal to the actual flow rate. The flow measurement uncertainty may be included in the BERR1 term in the CPC and is equal to or greater than 4%.

(8) - Above 70% of RATED THERMAL POWER, verify that the total RCS flow rate as indicated by each CPC is less than or equal to the actual RCS total flow rate determined by calorimetric calculations (conservatively compensated for measurement uncertainties).

INSTRUMENTATION

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 bypasses shown in Table 3.3-3 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3-4 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.2 The logic for the bypasses shall be demonstrated OPERABLE during the at power CHANNEL FUNCTIONAL TEST of channels affected by bypass operation. The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

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

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	~~7~~ 6
b. Containment Pressure - High	4	2	3	1, 2, 3	~~8*~~ 7*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	~~8*~~ 7*
d. Automatic Actuation - Logic	2	1	2	1, 2, 3	~~12~~ 9
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	~~7~~ 6
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	~~10~~ 7
c. Automatic Actuation Logic	2	1	2	1, 2, 3	~~12~~ 9
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	~~7~~ 6
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	~~7~~ 6
c. Containment Pressure - High	4	2	3	1, 2, 3	~~8*~~ 7*
~~d. Pressurizer Pressure - Low~~	~~4~~	~~2~~	~~3~~	~~1, 2, 3(a)~~	~~8*~~
d ~~e~~. Automatic Actuation Logic	2	1	2	1, 2, 3	~~12~~ 9

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
7. CONTAINMENT PURGE VALVES ISOLATION					
a. Manual (Purge Valve Control Switches)	2/Penetration	1/Penetration	2/Penetration	1, 2, 3, 4	7
b. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	7
c. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	7
d. Automatic CIAS Actuation Logic	2	1	2	1, 2, 3	12
e. Containment Radiation - High					
Gaseous Monitor	(1)	(1)	(1)	1, 2, 3, 4	9
Particulate Monitor	(1)	(1)	(1)	1, 2, 3, 4	9
Area Monitor	(1)	(1)	(1)	1, 2, 3, 4	9
5. ~~8.~~ CONTAINMENT COOLING (CCAS)					
a. Manual (Trip Buttons)	2 sets of 2	1 set of 2	2 sets of 2	1, 2, 3, 4	~~7~~ 6
b. Containment Pressure - High	4	2	3	1, 2, 3,	~~8~~ 7
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	~~8~~ 7
d. Automatic Actuation Logic	2	1	2	1, 2, 3	~~12~~ 9

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. MAIN STEAM LINE ISOLATION					
a. Manual (Trip Buttons	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3	~~11~~ 8
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	~~8*~~ 7*
c. Automatic Actuation Logic	2/steam generator	1/steam generator	2/steam generator	1, 2, 3	~~12~~ 9
5. SHIELD BUILDING FILTRATION (SBFAS)					
a. Manual SBFAS (Trip Buttons)	2	1	2	1, 2, 3, 4	7
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	7
c. Automatic Actuation Logic	2	1	2	1, 2, 3	12
d. Containment Pressure - High	4	2	3	1, 2, 3	8*
e. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	8*
f. Containment Radiation - High	4	2	3	1, 2, 3, 4	7
6. ~~CONTAINMENT SUMP~~ RECIRCULATION (~~S~~RAS)					
~~a. Manual SRAS (Trip Buttons)~~	~~2~~	~~1~~	~~2~~	~~1, 2, 3, 4~~	~~7~~
a. ~~b.~~ Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	~~8*~~ 7
b. ~~c.~~ Automatic Actuation Logic	2	1	2	1, 2, 3	~~12~~ 9

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
~~9. LOSS OF POWER~~					
~~a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)~~	~~4/Bus~~	~~2/Bus~~	~~3/Bus~~	~~1, 2, 3~~	~~7~~
~~b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)~~	~~4/Bus~~	~~2/Bus~~	~~3/Bus~~	~~1, 2, 3~~	~~7~~
7 ~~10.~~ EMERGENCY FEEDWATER (EFAS)					
a. Manual (Trip Buttons)	2 sets of 2 per S/G	1 set of 2 per S/G	2 sets of 2 per S/G	1, 2, 3	~~11~~ 8
b. Automatic Actuation Logic	2/SG	1/SG	2/SG	1, 2, 3	~~12~~ 9
c. SG Level and Pressure (A/B) - Low and Δ ~~WP~~ (A/B) - High	4/SG	2/SG	3/SG	1, 2, 3	~~8*~~ 7*
d. SG Level (A/B) - Low and No S/G Pressure - Low Trip (A/B)	4/SG	2/SG	3/SG	1, 2, 3, 4	9 ~~8*~~ 7*
~~e. Safety Injection~~	~~(See 1 above - Safety Injection Initiating Functions and Requirements)~~				

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than (400) psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to (500) psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below (600) psia; bypass shall be automatically removed at or above (600) psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

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

b. All functional units receiving an input from the tripped channel are also placed in the tripped condition within 1 hour.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 9 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge valves are maintained closed.

ACTION 10 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

ACTION 11 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 12 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

ACTION STATEMENTS

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

ACTION 7 - A. With The number of OPERABLE channels one less than the Total Number of Channels, the inoperable channel shall be placed in the bypass condition.

B. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE:

1. Place one inoperable channel in the bypassed condition and one inoperable channel in the tripped condition within 1 hours.

2. Operation may continue with one inoperable channel in the bypassed condition and one inoperable channel in the tripped condition provided all functional units receiving an input from the bypassed channel are also placed in the bypassed condition within 1 hour and all functional units receiving an input from the tripped inoperable channel are tripped.

C. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass all associated functional units as listed below.

	Process Measurement Circuit	Functional Unit Bypassed
1.	Containment Pressure - High	Containment Pressure - High (ESF) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low Steam Generator ΔP 1 and 2 (EFAS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

3. Steam Generator Level - Low — Steam Generator Level - Low; Steam Generator Level - High; Steam Generator ΔP (EFAS)

D. With an inoperable channel in the bypass condition, the channel shall be returned to the operable condition as soon as practical, but no longer than 90 days for failures outside containment or no longer than 18 months for failures inside containment, and the bypass removed.

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channels to OPERABLE status within 48 hours or be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in at least HOT SHUTDOWN within the following 6 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing provided the other channel is OPERABLE.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ ~~(18.4) psia~~ 2.95 psig	≤ ~~(19.02) psia~~ 3.14 psig
c. Pressurizer Pressure - Low	≥ ~~(1740)~~ 1806 psia (1)	≥ ~~(1686.7)~~ 1763 psia (1)
d. Automatic Actuation Logic	Not Applicable	Not Applicable
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ ~~(23.3) psia~~ 8.14 psig	≤ ~~(23.62) psia~~ 8.83 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ ~~(18.4) psia~~ 2.95 psig	≤ ~~(19.02) psia~~ 3.14 psig
~~d. Pressurizer Pressure - Low~~	~~≥ (1740) psia~~	~~≥ (1686.7) psia~~
d ~~e~~. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM ~~LINE~~ ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ ~~(500)~~ 729 psia (2)	≥ ~~(495)~~ 711 psia (2)
c. Automatic Actuation Logic	Not Applicable	Not Applicable

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. SHIELD BUILDING FILTRATION (SBFAS)		
a. Manual SBFAS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ (18.4) psia	≤ (19.02) psia
d. Pressurizer Pressure - Low	≥ (1740) psia	≥ (1686.7) psia
e. Containment Radiation - High		
1. Gaseous Monitor	≤ (2x background)	≤ (2x background)
2. Particulate Monitor	≤ (2x background)	≤ (2x background)
3. Area Monitor	≤ (2x background)	≤ (2x background)
f. Automatic Actuation Logic	Not Applicable	Not Applicable
6. ~~CONTAINMENT SUMP~~ RECIRCULATION (~~S~~RAS)		
a. Manual ~~S~~RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	~~(30) inches above tank bottom~~ 18.5% TAP SPAN	~~(24) inches above tank bottom~~ BETWEEN 17.74% TAP SPAN And 19.26% TAP SPAN
c. Automatic Actuation Logic	Not Applicable	Not Applicable
7. CONTAINMENT PURGE VALVES ISOLATION (CPAS)		
a. Manual (Purge Valve Control Switches)	Not Applicable	Not Applicable
b. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
d. Automatic CIAS Actuation Logic	Not Applicable	Not Applicable
e. Containment Radiation - High		
1. Gaseous Monitor	≤ (2x background)	≤ (2x background)
2. Particulate Monitor	≤ (2x background)	≤ (2x background)
3. Area Monitor	≤ (2x background)	≤ (2x background)

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT		TRIP VALUE	ALLOWABLE VALUES
5 ~~8~~.	CONTAINMENT COOLING (CCAS)		
a.	Manual (Trip Buttons)	Not Applicable	Not Applicable
b.	Containment Pressure - High	< ~~(18.4) psia~~ 2.95 Psig	< ~~(19.02) psia~~ 3.14 Psig
c.	Pressurizer Pressure - Low	> ~~(1740)~~ 1806 psia (1)	> ~~(1686.7)~~ 1763 psia (1)
d.	Automatic Actuation Logic	Not Applicable	Not Applicable
~~9.~~	~~LOSS OF POWER~~		
~~a.~~	~~4.16 kv Emergency Bus Undervoltage (Loss of Voltage)~~	~~(3120) volts~~	~~(3120) volts~~
~~b.~~	~~4.16 kv Emergency Bus Undervoltage (Degraded Voltage)~~	~~(423) ± (2.0) volts with an (8.0 ± 0.5) second time delay~~	~~(423) ± (4.0) volts with an (8.0 ± 0.8) second time delay~~
7. ~~10~~.	EMERGENCY FEEDWATER (EFAS)		
a.	Manual (Trip Buttons)	Not Applicable	Not Applicable
b.	Steam Generator (A&B) Level-Low	> ~~(46.5)%~~ 23% (3)	> ~~(45.61)%~~ 22.23% (3)
c.	Steam Generator ΔP-High (SG-A > SG-B)	< ~~(39)~~ 50 psi	< ~~(48.35)~~ 66.25 psi
d.	Steam Generator ΔP-High (SG-B > SG-A)	< ~~(39)~~ 50 psi	< ~~(48.35)~~ 66.25 psi
e.	Steam Generator (A&B) Pressure - Low	> ~~(728)~~ 729 psia (2)	> ~~(706.6)~~ 711 psia (2)
~~f.~~	~~Safety Injection~~	~~See 1 above (All S.I. Setpoints)~~	
f ~~g~~.	Automatic Actuation Logic	Not Applicable	Not Applicable

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TABLE NOTATION

(1) Value may be decreased manually, to a minimum of greater than or equal to 300 psia, as pressurizer pressure is reduced, provided the margin between the pressurizer and this value is maintained at less than or equal to 400 psia;* the setpoint shall be increased automatically as pressurizer pressure is increased until the trip setpoint is reached. Trip may be manually bypassed below 400 psia; bypass shall be automatically removed whenever pressurizer is greater than or equal to 400 psia.

(2) Value may be decreased manually as steam generator pressure is reduced, provided the margin between the steam generator pressure and this value is maintained at less than or equal to 200 psi;* the setpoint shall be increased automatically as steam generator pressure is increased until the trip setpoint is reached.

(3) % of the distance between steam generator upper and lower level instrument nozzles.

*Variable setpoints are for use only during <u>normal</u>, <u>controlled</u> plant heatups and cooldowns.

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection ~~(ECCS)~~	Not Applicable
~~Containment Isolation~~	~~Not Applicable~~
~~Shield Building Filtration System~~	~~Not Applicable~~
~~Containment Purge Valve Isolation~~	~~Not Applicable~~
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
~~Shield Building Filtration System~~	~~Not Applicable~~
~~Containment Purge Valve Isolation~~	~~Not Applicable~~
d. MSIS	
Main Steam Isolation	Not Applicable
~~e. SBFAS~~	
~~Shield Building Filtration System~~	~~Not Applicable~~
f. ~~S~~RAS	
Containment Sump Recirculation	Not Applicable
~~g. CPAS~~	
~~Containment Purge Valve Isolation~~	~~Not Applicable~~
e. ~~h.~~ CCAS	
Containment Cooling	Not Applicable
i. EFAS	
~~Emergency Feedwater Pumps~~	~~Not Applicable~~
TRAIN A	Not Applicable
TRAIN B	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure-Low	
a. Safety Injection ~~(ECCS)~~	~~≤ ___*/___**~~
1. ~~b. Containment Isolation~~ High Pressure Safety Injection	≤ 26.5*/15.5**
2. ~~c. Shield Building Filtration System~~ Low Pressure Safety Injection	≤ LATER*/LATER**
~~d. Containment Cooling~~	~~≤ ___*/___**~~
3. Containment Pressure-High	
1. High Pressure Safety Injection	≤ 26.5*/15.5**
2. Low Pressure Safety Injection	LATER / LATER
a. Safety Injection ~~(ECCS)~~	~~≤ ___*/___**~~
b. Containment Isolation	≤ LATER*/LATER**
~~c. Shield Building Filtration System~~	~~≤ ___*/___**~~
c ~~d~~. Containment Cooling	≤ 31*/17**
4. Containment Pressure--High-High	
a. Containment Spray	≤ 30.3*/26.3**
~~5. Containment Radiation-High~~	
~~a. Containment Purge Valves Isolation~~	~~≤ ___/___**~~
~~b. Shield Building Filtration System~~	~~≤ ___*/___**~~
5 ~~6~~. Steam Generator Pressure-Low	
a. Main Steam Isolation	≤ 5.5 / ~~___**~~
b. Emergency Feedwater	≤ 20.5/ ~~___**~~
6 ~~7~~. Refueling Water Storage Tank-Low	
a. Containment Sump ~~Recirculation~~ VALVE OPEN	≤ LATER / ~~___**~~
~~8. 4.16 Kv Emergency Bus Undervoltage (Loss of Voltage)~~	
~~a. Loss of Power~~	~~≤ ___~~
~~9. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)~~	
~~a. Loss of Power~~	~~≤ ___~~

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
7 ~~10~~. Steam Generator Level-Low	
a. Emergency Feedwater-TRAIN A	LATER ~~≤ */ **~~
b Emergency Feedwater-TRAIN B	LATER
8 ~~11~~. Steam Generator ΔP-High-Coincident With Steam Generator Level Low	
a. Emergency Feedwater-TRAIN A	LATER ~~≤ */ **~~
b. Emergency Feedwater-TRAIN B	LATER

NOTE: Response time for Motor-Driven Auxiliary Feedwater Pumps on all S.I. signal starts ≤ (60.0)

TABLE NOTATION

*Diesel generator starting and sequence loading delays included. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

**Diesel generator starting and sequence loading delays not included. Offsite power available. Response time limit includes movement of valves and attainment of pump or blower discharge pressure.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M(2)	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure -- High - High	S	R	M(2)	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Containment Pressure - High	S	R	M(2)	1, 2, 3
~~d. Pressurizer Pressure - Low~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3~~
d. ~~e.~~ Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
4. MAIN STEAM ~~LINE~~ ISOLATION (MSIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Steam Generator Pressure - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
~~5. SHIELD BUILDING FILTRATION (SBFAS)~~				
~~a. Manual SBFAS (Trip Buttons)~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~
~~b. Manual SIAS (Trip Buttons)~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~
~~c. Containment Pressure - High~~	~~S~~	~~R~~	~~M(2)~~	~~1, 2, 3~~
~~d. Pressurizer Pressure - Low~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3~~
~~e. Containment Radiation - High~~				
~~Gaseous Monitor~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
~~Particulate Monitor~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
~~Area Monitor~~	~~S~~	~~R~~	~~M~~	~~1, 2, 3, 4~~
~~f. Automatic Actuation Logic~~	~~N.A.~~	~~N.A~~	~~M(1)~~	~~1, 2, 3~~

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. ~~CONTAINMENT SUMP~~ RECIRCULATION (~~S~~RAS)				
~~a. Manual SRAS (Trip Buttons)~~	~~N.A.~~	~~N.A.~~	~~R~~	~~N.A.~~
a. ~~b.~~ Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
b. ~~c.~~ Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
7. CONTAINMENT PURGE VALVES ISOLATION				
a. Manual (Purge Valve Control Switches)	N.A.	N.A.	R	N.A.
b. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
d. Automatic CIAS Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
e. Containment Radiation - High				
Gaseous Monitor	S	R	M	1, 2, 3, 4
Particulate Monitor	S	R	M	1, 2, 3, 4
Area Monitor	S	R	M	1, 2, 3, 4
5. ~~8.~~ CONTAINMENT COOLING (CCAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Containment Pressure - High	S	R	M(2)	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
9. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	S	R	R	1, 2, 3
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	S	R	R	1, 2, 3

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TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
7 ~~10.~~ EMERGENCY FEEDWATER (EFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. SG Level and Pressure (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
~~e. S.I.~~	~~(See 1 above (S.I. Surveillance Requirements)~~			

TABLE NOTATION

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

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

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 CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-3.

TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

(SEE INSERT "A", ATTACHED)

Delete

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area					
i. Criticality Monitor	(1)	*	$\leq$ 15 mR/hr	$(10^{-1} - 10^{4})$ mR/hr	13
ii. Ventilation System Isolation	(1)	**	($\leq$ 2x background)	$(1 - 10^{5})$ cpm	15
b. Containment - Purge & Exhaust Isolation	(1)	6	($\leq$ 2x background)	$(1 - 10^{5})$ cpm	16
c. Control Room Isolation	(1)	ALL MODES	($\leq$ 2x background)	$(10^{-1} - 10^{4})$ mR/hr	17
2. PROCESS MONITORS					
a. Fuel Storage Pool Area Ventilation System Isolation					
i. Gaseous Activity	(1)	**	($\leq$ 2x background)	$(1 - 10^{5})$ cpm	15
ii. Particulate Activity	(1)	**	($\leq$ 2x background)	$(1 - 10^{5})$ cpm	15
b. Containment					
i. Gaseous Activity					
a) Purge & Exhaust Isolation	(1)	6	($\leq$ 2x background)	$(1 - 10^{5})$ cpm	16
b) RCS Leakage Detection	(1)	1, 2, 3 & 4	Not Applicable	$(1 - 10^{5})$ cpm	14
ii. Particulate Activity					
a) Purge & Exhaust Isolation	(1)	6	($\leq$ 2x background)	$(1 - 10^{5})$ cpm	16
b) RCS Leakage Detection	(1)	1, 2, 3 & 4	Not Applicable	$(1 - 10^{5})$ cpm	14

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

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TABLE 3.3-6

RADIATION MONITORING INSTRUMENTATION

INSERT "A"

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Containment - Purge & Exhaust Isolation	(1)	6	(≤ 2x background)	10^{-1} - 10^{5} mR/hr	15
2. PROCESS MONITORS					
a. Fuel Storage Pool Area Ventilation System Isolation					
i. Gaseous	(1)	*	(≤ 2x background)	10^{1} - 10^{7} cpm	14
ii. Particulate/Iodine	(1)	*	(≤ 2x background)	10^{1} - 10^{7} cpm	14
b. Containment					
i. Gaseous					
a) Purge & Exhaust Isolation	(1)	6	(≤ 2x background)	10^{1} - 10^{7} cpm	15
b) RCS Leakage Detection	(1)	1, 2, 3 & 4	Not Applicable	10^{1} - 10^{7} cpm	13
ii. Particulate					
a) Purge & Exhaust Isolation	(1)	6	(≤ 2x background)	10^{1} - 10^{7} cpm	15
b) RCS Leakage Detection	(1)	1, 2, 3 & 4	Not Applicable	10^{1} - 10^{7} cpm	13
iii. Iodine					
a) Purge & Exhaust Isolation	(1)	6	(≤ 2x background)	10^{1} - 10^{7} cpm	15
b) RCS Leakage Detection	(1)	1, 2, 3 & 4	Not Applicable	10^{1} - 10^{7} cpm	13
c. Control Room Isolation					
i. Particulate/Iodine	(1)	ALL MODES	(≤ 2x background)	10^{1} - 10^{7} cpm	16
ii. Gas	(1)	ALL MODES	(≤ 2x background)	10^{1} - 10^{7} cpm	16

* With irradiated fuel in the storage pool.

TABLE 3.3-6 (Continued)

TABLE NOTATION

~~ACTION 13 - 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 ~~14~~ 13 - 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 ~~15~~ 14 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.~~12~~ 11).

ACTION ~~16~~ 15 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification (3.9.~~9~~ 8).

ACTION ~~17~~ 16 - 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 recirculation mode of operation.

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~~CE STS~~ SAN ONOFRE - UNIT 2

TABLE 4.3-3 [SEE INSERT "A", ATTACHED]

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

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INSERT "A"

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Containment - Purge & Exhaust Isolation	S	R	M	6
2. PROCESS MONITORS				
a. Fuel Storage Pool Area - Ventilation System Isolation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
b. Containment				
i. Gaseous				
a) Purge & Exhaust Isolation	S	R	M	6
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate				
a) Purge & Exhaust Isolation	S	R	M	6
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
iii. Iodine				
a) Purge & Exhaust Isolation	S	R	M	6
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
c. Control Room Isolation				
i. Particulate/Iodine	S	R	M	ALL MODES
ii. GAS	S	R	M	ALL MODES

* With irradiated fuel in the storage pool.

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

a. At least 75% of all incore detector locations, and

b. A minimum of two quadrant symmetric incore detector locations per core quadrant.

An OPERABLE incore detector location shall consist of a fuel assembly containing a fixed detector string with a minimum of four OPERABLE rhodium detectors or an OPERABLE movable incore detector capable of mapping the location.

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

a. AZIMUTHAL POWER TILT,

b. Radial Peaking Factors,

c. Local Power Density,

d. DNB Margin.

ACTION:

With the 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 incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for monitoring the AZIMUTHAL POWER TILT, radial peaking factors, local power density or DNB margin:

b. At least once per 18 months by performance of a CHANNEL CALIBRATION operation which exempts the neutron detectors but includes all electronic components. The neutron detectors shall be calibrated prior to installation in the reactor core.

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INSTRUMENTATION

SEISMIC INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

Instruments & Sensor Locations	Measurement Range	Minimum Instruments Operable
1. Triaxial Time-History Strong Motion Accelerometers		
a. Steam Generator Base Support	-2 to +2g	1
b. Pressurizer Base Support	-2 to +2g	1
c. Top of Reactor Coolant Pump Motor	-2 to +2g	1
d. Containment base in Tendon Gallery	-2 to +2g	1
e. Containment Operating Level	-2 to +2g	1
f. Unit #1 Free Field	-1 to +1g	1
g. Control Building Basement	-2 to +2g	1
h. Control Building Roof	-2 to +2g	1
i. Safety Equipment Building Base Slab	-2 to +2g	1
j. Safety Equipment Building Piping Support	-2 to +2g	1
k. Radwaste Building Equipment Support	-2 to +2g	1
2. Triaxial Peak Recording Accelerographs		
a. Control Building-Control Room	-2 to +2g	1
b. Control Building Base	-2 to +2g	1
c. Top of Containment Structure	-5 to +5g	1
d. Reactor Coolant Piping	-2 to +2g	1
3. Seismic Triggers		
a. Containment Base in Tendon Gallery	+0.005 to +0.05g	1
b. Containment Operating Level	+0.005 to +0.05g	1
4. Seismic Switches		
a. Steam Generator Base Support	Set pt. 0.45 Horz/0.30 Vert.	1**
b. Containment Base in Tendon Gallery	Set pt. 0.40 Horz/0.50 Vert.	1**
5. Seismic Alarm Annunciator (4a & 4b are sensors)		
a. Control Room Panel L-167	-	1
6. Peak Shcok Recorder		
a. Containment Base in Tendon Gallery	2 to 25.4 Hz 1.6 to 90g	1**
7. Peak Shock Annunicator	2 to 25.4 Hz 1.6 to 90g	1
a. Control Room Panel L-167		

** With control room indication

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TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS & SENSOR LOCATIONS	CHANNEL CHECK	FUNCTIONAL CALIBRATION	CHANNEL CHANNEL TEST
1. Triaxial Time-History Strong Motion Accelerometers			
a. Steam Generator Base Support	M*	R	SA
b. Pressurizer Base Support	M*	R	SA
c. Top of Reactor Coolant Pump Motor	M*	R	SA
d. Containment Base in Tendon Gallery	M*	R	SA
e. Containment Operating level	M*	R	SA
f. Unit #1 Free Field	M*	R	SA
g. Control Building Basement	M*	R	SA
h. Control Building Roof	M*	R	SA
i. Safety Equipment Building Base SLAB	M*	R	SA
j. Safety Equipment Building Piping Support	M*	R	SA
k. Radwaste Building Equipment Support	M*	R	SA
2. Triaxial Peak Recording Accelerographs			
a. Control Building-Control Room	N/A	R	N/A
b. Control Building Base	N/A	R	N/A
c. Top of Containment Structure	N/A	R	N/A
d. Reactor Coolant Piping	N/A	R	N/A
3. Seismic Triggers			
a. Containment Base in Tendon Gallery	M	R	SA
b. Containment Operating level	M	R	SA
4. Seismic Switches			
a. Steam Generator Base Support	M	R**	SA**
b. Containment Base in Tendon Gallery	m	R**	SA**
5. Seismic Alarm Annunciators (4a & 4b are sensors)			
a. Control Room Panel L-167	M	R	SA
6. Peak Shock Recorder			
a. Containment Base in Tendon Gallery	N/A	R**	N/A
7. Peak Shock Annunciator			
a. Control Room Panel L-167	N/A	R**	N/A

(* Except seismic trigger)
(** With Control Room indication)

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

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		
a. 0-50 MPH,	Nominal Elev. 10 METERS	1
b. 0-100 MPH,	Nominal Elev. 20 METERS	1
2. WIND DIRECTION		
a. 0-360-180°,	Nominal Elev. 10 METERS	1
b. 0-360-180°,	Nominal Elev. 20 METERS	1
3. AIR TEMPERATURE ~~- DELTA T~~		
a. -30 TO +50°C,	Nominal Elev. 10 METERS	1
~~b. ______,~~	~~Nominal Elev. ______~~	~~1~~
4. Delta Temperature		
a. -3°C To +3°C	Nominal Elev. 10/40 METERS	1
b. -3°C To +3°C	Nominal Elev. 10/40 METERS	1
5. Sigma Azimuth		
a. -0 To 45°	Nominal Elev 10 METERS	1

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 10 METERS	D	SA
b. Nominal Elev. 20 METERS	D	SA
c. Nominal Elev 40 METERS	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 10 METERS	D	SA
b. Nominal Elev. 20 METERS	D	SA
c. Nominal Elev 40 METERS	D	SA
3. AIR TEMPERATURE ~~DELTA T~~		
a. Nominal Elev. 10 METERS	D	SA
~~b. Nominal Elev.~~	~~D~~	~~SA~~
4. Delta Temperature		
a. Nominal Elev 10/40 METERS	D	SA
b. Nominal Elev 10/40 METERS	D	SA
5. Sigma Azimuth		
a Nominal Elev. 10 METERS	D	SA

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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, either 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

LOCATED IN PANEL L042

INSTRUMENT	~~READOUT LOCATION~~ CHANNELS	CHANNELS RANGE	MINIMUM CHANNELS OPERABLE
1. Log Power Level ~~Power Range Neutron Flux~~	2	10^{-8} To 200%	1
2. Reactor Coolant Boron Concentration ~~Intermediate Range Neutron Flux~~	1	0 To 2500 PPM	1
3. Condensate Storage Tank Level ~~Source Range Neutron Flux~~	1/Tank	0 To 100%	1
4. Condenser Vacum ~~Reactor Trip Breaker Indication~~	1/LP Zone	0 To 5" Hg ~~OPEN-CLOSE~~	1 ~~1/trip breaker~~
5. Reactor Coolant Cold Leg Temperature - ~~Average~~	1/Loop	0 To 600°F	1
6. Volume Control Tank Level ~~Reactor Coolant Flow Rate~~	1	0 To 100%	1
7. Pressurizer Pressure	2	0 To 3000 PSIA	1
8. Pressurizer Level	2	0 To 100%	1
9. Steam Generator Pressure	1/Steam Generator	0 To 1200 PSIA	1/steam generator
10. Steam Generator Level	1/Steam Generator	0 To 100%	1/steam generator
11. ~~Control Rod Position Limit Switches~~ Letdown H-X Pressure	1	0 To 600 PSIG	1 ~~1 insertion limit switch/rod~~
12. Letdown H-X Temperature ~~Shutdown Cooling Flow Rate~~	1	0 To 200°F	1
13. Boric Acid Makeup Tank Level ~~Shutdown Cooling Temperature~~	1/Tank	0 To 100%	1
~~14. Auxiliary Feedwater Flow Rate~~			~~1~~

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TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Log Power Level ~~Power Range Neutron Flux~~	M	~~Q~~ R
2. Reactor Coolant Boron Concentration ~~Intermediate Range Neutron Flux~~	M	~~N.A.~~ R
3. ~~Source Range Neutron Flux~~	M	~~N.A.~~
4. ~~Reactor Trip Breaker Indication~~	M	~~N.A.~~
5. Reactor Coolant Cold Leg Temperature-Average	M	R
6. Volume Control Tank Level ~~Reactor Coolant Flow Rate~~	M	R
7. Pressurizer Pressure	M	R
8. Pressurizer Level	M	R
9. Steam Generator Level	M	R
10. Steam Generator Pressure	M	R
11. Letdown H-x Pressure ~~Control Rod Position Limit Switches~~	M	R
12. Letdown H-X Temperature ~~Shutdown Cooling Flow Rate~~	M	R
13. Boric Acid Makeup Tank Level ~~Shutdown Cooling Temperature~~	M	R
~~14. Auxiliary Feedwater Flow Rate~~	~~M~~	~~R~~

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Total Number of channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in 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 channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION (LATER)

INSTRUMENT (Illustrational Only)	MINIMUM CHANNELS OPERABLE	TOTAL NUMBER OF CHANNELS
1. Containment Pressure	1	2
2. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	1	2
3. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	1	2
4. Reactor Coolant Pressure - Wide Range	1	2
5. Pressurizer Water Level	1	2
6. Steam Line Pressure	1/steam generator	2/steam generator
7. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level - Wide Range	1/steam generator	1/steam generator
9. Refueling Water Storage Tank Water Level	1	2
10. Boric Acid Storage Tank Solution Level	1	2
11. Auxiliary Feedwater Flow Rate	1	2
12. Reactor Cooling System Subcooling Margin Monitor	1	2
13. PORV Position Indicator	1/valve	2/valve
14. PORV Block Valve Position Indicator	1/valve	2/valve
15. Safety Valve Position Indicator	1/valve	2/valve

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TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(LATER)

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

N/A

INSTRUMENTATION

CHLORINE DETECTION SYSTEMS (OPTIONAL)

LIMITING CONDITION FOR OPERTION

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

APPLICABILITY: ALL MODES

ACTION:

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

b. With no chlorine detection system OPERABLE, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

c. The provisions of Specification 3.0.4 are not applicable.

d. The provisions of Specification 3.0.3 are not applicable in MODE 6.

SURVEILLANCE REQUIREMENTS

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

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the number of OPERABLE fire detection instrument(s) less than the minimum number OPERABLE requirement of Table 3.3-11:

a. Within 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect the containment at least once per 8 hours or (monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.6).

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.8.3 The non-supervised circuits associated with detector alarms between the instruments and the control room shall be demonstrated OPERABLE at least once per 31 days.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

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

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

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

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3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours either restore the system to OPERABLE status or isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

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

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

1. (Four) high pressure turbine stop valves.
2. (Four) high pressure turbine control valves.
3. (Four) low pressure turbine reheat stop valves.
4. (Four) low pressure turbine reheat intercept valves.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.
AND ASSOCIATED STEAM GENERATOR,

APPLICABILITY: 1 and 2.*

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

4.4.1.1 No additional Surveillance Requirements other than those required by Specification 4.3.1.1.

*See Special Test Exception 3.10.3.

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REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The reactor coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop ~~(A)~~ 1, and at least one associated reactor coolant pump.

its Associated STEAMGENERATOR,

2. Reactor coolant Loop ~~(B)~~ 2, and at least one associated reactor coolant pump.

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

APPLICABILITY: MODE 3

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

2. Reactor coolant Loop ~~(B)~~ 2 and its associated steam generator and at least one associated reactor coolant pump, ~~**~~

3. Shutdown Cooling ~~Loop~~ Train {A}#

4. Shutdown Cooling ~~Loop~~ Train {B}#

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

APPLICABILITY: MODES 4 and 5

ACTION:

a. With less than the above required coolant loops OPERABLE, immediately initiate corective action to return the required coolant loops to OPERABLE status as soon as possible, be in COLD SHUTDOWN within 20 hours.

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

*All reactor coolant pumps and ~~decay heat~~ shutdown cooling 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.

~~**A reactor coolant pump shall not be started with one or more of the RCS cold leg temperatures less than or equal to (275)°F unless 1) the pressurizer water volume is less than (300) cubic feet or 2) the secondary water temperature of each steam generator is less than (46)°F above each of the RCS cold leg temperatures.~~

#The normal or emergency power source may be inoperable in MODE 5.

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SURVEILLANCE REQUIREMENTS

4.4.1.3.1 The required shutdown cooling loop(s) shall be determined OPERABLE per Specification 4.0.5.

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

4.4.1.3.3 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be ≥ (LATER) at least once per 12 hours.

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

3/4.4.2 SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.*

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.4.3 SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.*

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.4.4 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with at least (LATER) Kw of pressurizer heaters a water volume of less than or equal to (LATER) cubic feet.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the pressurizer inoperable due to an inoperable emergency power supply to the pressurizer heaters either restore the inoperable emergency power supply within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

3/4.4.5 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.5.1 Each PORV shall be demonstrated OPERABLE:

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

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

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

(4.4.5.3 The emergency power supply for the PORVs and block valve shall be demonstrated OPERABLE at least once per 18 months by transferring motive and control power from the normal to the emergency power supply and operating the valves through a through a complete cycle of full travel.)

3/4.4.~~8~~5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

(LATER)

3.4.~~8~~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.~~8~~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.~~8~~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.~~8~~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.~~8~~5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.~~8~~5.4. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

3. A tube inspection (pursuant to Specification 4.4.~~6~~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.

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SURVEILLANCE REQUIREMENTS (Continued)

4.4.~~6~~5.3 Inspection Frequencies - The above required inservice inspections of steam generator tubes shall be performed at the following frequencies:

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

b. If the results of the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 at 40 month intervals fall into Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.~~6~~5.3.a; the interval may then be extended to a maximum of once per 40 months.

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

1. Primary-to-secondary tubes leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.~~6~~5.2.

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.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.~~6~~5.4 Acceptance Criteria

a. As used in this Specification

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

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

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

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

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

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

7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.6.3.c, above.

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

9. 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.

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

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SURVEILLANCE REQUIREMENTS (Continued)

condition of the tubing. This inspection shall be performed after the field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

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

4.4.~~6~~5.5 Reports

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

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

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

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

TABLE 4.4-1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

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

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

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TABLE 4.4–2

STEAM GENERATOR TUBE INSPECTION

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

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

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3/4.4.~~7~~6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.~~7~~6.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

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

b. The containment sump ~~level and~~ inlet flow monitoring system~~, and~~

~~c. Either the (containment air cooler condensate flow rate) or a containment atmosphere (gaseous or particulate) radioactivity monitoring system.~~

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days, ~~provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactivity 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.~~7~~6.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

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

~~c. (Specify appropriate surveillance tests depending upon the method of leakage detection system selected.)~~

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.~~7~~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. __ gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.~~

(LATER)

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

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

b. With any Reactor Coolant System leakage greater than any one of the 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. SEE INSERT "A" ATTACHED

SURVEILLANCE REQUIREMENTS

4.4.~~7~~6.2.a Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

1 ~~a~~. Monitoring the containment atmosphere (gaseous or particulate) radioactivity monitor at least once per 12 hours.

2 ~~b~~. Monitoring the containment sump inventory and inlet flow ~~discharge~~ at least once per 12 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.~~

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INSERT "A"

(LATER)

~~2.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 (Continued)

~~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,~~

c. ~~d.~~ Performance of a Reactor Coolant System water inventory balance at least once per 72 hours. ~~during steady state operation, and~~. The provisions of Specification 4.0.4 are not applicable for Entry into MODE 4.

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

4.4.6 ~~7~~.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5 except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying Reactor Coolant System leakage to be within its limit.

a. At least once per 18 months

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

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

(LATER)

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REACTOR COOLANT SYSTEM

3/4.4.~~8~~7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.~~8~~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.

MODES 5 and 6

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 psia, 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 psia or prior to proceeding to MODE 4.

SURVEILLANCE REQUIREMENTS

4.4.~~8~~7 The Reactor Coolant System chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.4-3.

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

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

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

☆☆ In Mode 6, with the RCS open to the atmosphere, the concentration shall not exceed 0.40 ppm for a period of more than seven (7) consecutive days.

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TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

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

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

3/4.4.~~9~~8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.~~9~~8 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to ~~1.0~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131, and

b. Less than or equal to 100/$\bar{E}$ microcuries/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~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131 but within the allowable limit (below and to the left of the line) shown on Figure 3.4-1, operation may continue for up to 48 hours provided that the cumulative operating time under, in modes 1, 2 and 3, these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time, in modes 1, 2 and 3, at a primary coolant specific activity greater than ~~1.0~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131 exceeding 500 hours in any consecutive 6 month period, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days indicating the number of hours above this limit. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant greater than ~~1.0~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

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

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

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ACTION: (Continued)

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant greater than 6.5 ~~1.0~~ microcurie/gram DOSE EQUIVALENT I-131 or greater than 100/Ē microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

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

2. Fuel burnup by core region,

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

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

5. The time duration when the specific activity of the primary coolant exceeded ~~1.0~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.~~8~~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.

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SAN ONOFRE - UNIT 2

~~CE-STS~~

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 6.5 ~~1.0~~ μCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ μCi/gram, and	1#, 2#, 3#, 4#, 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

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

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

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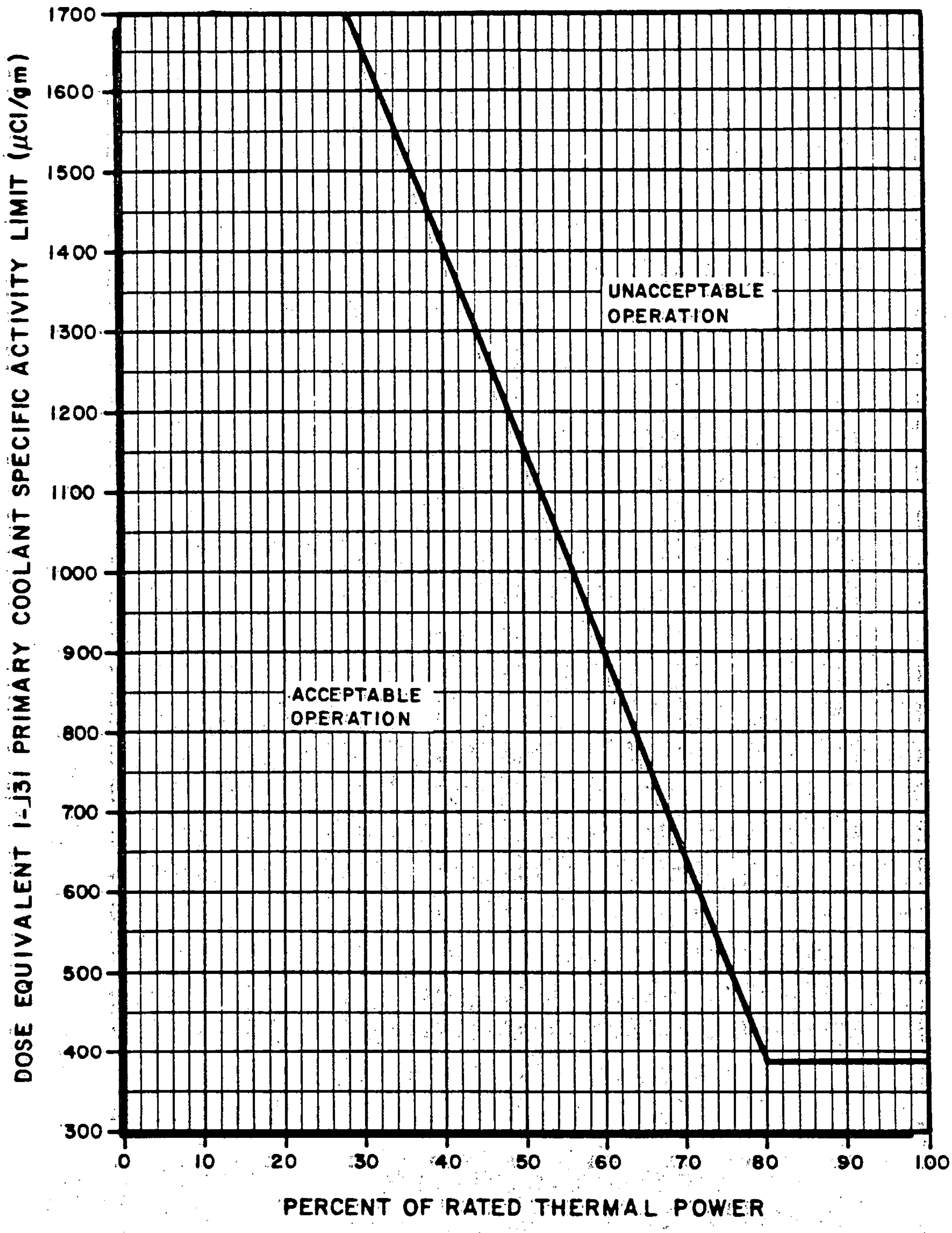

Figure 3.4-1

DOSE EQUIVALENT I-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY 6.5 Ci/gram DOSE EQUIVALENT I-131

REACTOR COOLANT SYSTEM

3/4.4.~~10~~ 9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.~~10~~ 9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figure {3.4-2} during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of ~~{100}~~ 60°F in any one hour period.

b. A maximum cooldown of {100}°F in any one hour period ~~for~~ for RCS temperatures above 200°F; a maximum cooldown of 60°F in any one hour period for temperatures below 200°F.

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

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.~~10~~ 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.~~10~~ 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 Figure 3.4-2. amended as of April 19, 1976,

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FIGURE 3.4-2

(LATER)

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.~~10~~ 9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of [200]°F in any one hour period,

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

c. A maximum ~~spray water temperature differential of [200]°F.~~ of 1000 spray cycles with a spray water temperature differential of greater than 200 °F.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.~~10~~ 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.

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REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.~~10~~9.3 At least one of the following overpressure protection systems shall be OPERABLE:

a. The Shutdown Cooling System (SDCS) Relief Valve (PSV9349) ~~Two power operated relief valves (PORVs)~~ with a lift setting of less than or equal to ~~{450}~~ 417 psig, or

b. A reactor coolant system vent of greater than or equal to (LATER) ~~(1.3)~~ square inches.

APPLICABILITY: When the temperature of one or more of the RCS cold legs is less than or equal to (LATER) ~~(275)°F~~, except when the reactor vessel head is removed.

ACTION:

a. With the SDCS Relief Valve ~~one PORV~~ inoperable, ~~either restore the inoperable PORV to OPERABLE status within 7 days or~~ depressurize and vent the RCS through a greater than or equal to (LATER) ~~(1.3)~~ square inch vent~~(s)~~ within the next 8 hours; maintain the RCS in a vented condition until ~~both the PORVs have~~ SDCS Relief Valve has been restored to OPERABLE status.

~~b. With both PORVs inoperable, depressurize and vent the RCS through a greater than or equal to (1.3) square inch vent(s) within 8 hours; maintain the RCS in a vented condition until both PORVs have been restored to OPERABLE status.~~

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

c ~~d~~. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.~~10~~9.3.1 Each SDCS Relief Valve ~~PORV~~ shall be demonstrated OPERABLE by:

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

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SURVEILLANCE REQUIREMENTS (Continued)

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

a ~~c~~. Verifying the SDCS Relief Valve ~~PORV~~ isolation valves are ~~is~~ open at least once per 72 hours* when the SDCS Relief Valve ~~PORV~~ is being used for overpressure protection.

~~b d. Testing pursuant to Specification 4.0.5. Inservice test intervals shall not exceed 30 months~~

4.4.9 ~~10~~.3.2 The RCS vent~~(s)~~ shall be verified to be open at least once per 12 hours* when the vent~~(s)~~ is being used for overpressure protection.

b. Testing in accordance with the inservice test requirements for ASME Category C valves pursuant to Specification 4.0.5. Inservice test intervals shall not exceed 30 months.

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

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REACTOR COOLANT SYSTEM

~~3.4.11~~ 3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 AND 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

~~3.4.11~~ 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 limits 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 to within its limit or isolate the affected component from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

~~4.4.11~~ 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.

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3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between ~~(1413)~~ 1680 and ~~(1539)~~ 1807 cubic feet,

c. Between ~~(1731)~~ 1720 and ~~(2250)~~ 2300 ppm of boron, and

d. A nitrogen cover-pressure of between {600} and ~~(624)~~ (LATER) psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank 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 safety injection tank 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 be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying (by the absence of alarms) the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each safety injection tank isolation valve is open.

*With pressurizer pressure greater than or equal to {700} psia.

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SURVEILLANCE REQUIREMENTS (Continued)

b. ~~At least once per 31 days and w~~ 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 safety injection tank solution.

c. At least once per 31 days when the RCS pressure is above {2000} psia, by verifying that power to the isolation valve operator is disconnected ~~by removing the breaker from the circuit.~~ with circuit breaker padlocked in the open position

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

   1. Before the RCS pressure increases above 505 psia, and ~~When the RCS pressure exceeds (700) psia, and~~

   2. Upon receipt of a safety injection test signal.

4.5.1.2 Each safety injection tank water level and pressure channel shall be demonstrated OPERABLE:

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

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

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 350 ~~300~~°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE high-pressure safety injection pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a ~~Sump~~ Recirculation Actuation Signal.

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.

*With pressurizer pressure greater than or equal to 1700 psia.

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:~~

SEE INSERT "A" ATTACHED

~~Valve Number~~	~~Valve Function~~	~~Valve Position~~
~~a. ____~~	~~a. ____~~	~~a. ____~~
~~b. ____~~	~~b. ____~~	~~b. ____~~
~~c. ____~~	~~c. ____~~	~~c. ____~~

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

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

1. For all accessible areas of the containment prior to establishing CONTAINMENT INTEGRITY~~, and~~ following a COLD SHUTDOWN, and

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

d. At least once per 18 months by:

1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System ~~when the Reactor Coolant System pressure is above (300) psia.~~ to prevent opening of the Shutdown Cooling System isolation valves when RCS pressure ≥ 376 psia.

INSERT "A"

	Valve Number	Valve Function	Valve Position
a.	HV 9300	RWST Supply to ECCS Pumps	Open
b.	HV 9301	RWST Supply to ECCS Pumps	Open
c.	HV 9316	SDC Flow Bypass Control	Closed
d.	HV 9420	Hot Leg Injection Isolation	Closed
e.	HV 9434	Hot Leg Injection Isolation	Closed

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.

~~3. Verifying that a minimum total of (65) cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.~~

~~4. Verifying that when a representative sample of ____ ± ____ lbs of TSP from a TSP storage basket is submerged, without agitation, in ____ ± ____ gallons of ____ ± ____ °F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 6 within 4 hours.~~

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 SIAS and RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

   a. High-Pressure Safety Injection pump.

   b. Low-Pressure Safety Injection pump.

3. Verifying that on a ~~Sump~~ Recirculation Actuation Test Signal, the containment sump isolation valves open and the recirculation valve to the refueling water tank closed.

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

1. High-Pressure Safety Injection pump greater than or equal to LATER psig.

2. Low-Pressure Safety Injection pump greater than or equal to LATER psig.

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

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

~~2. At least once per 18 months.~~

~~HPSI System Valve Number~~	~~LPSI System Valve Number~~
~~a. ____~~	~~a. ____~~
~~b. ____~~	~~b. ____~~
~~c. ____~~	~~c. ____~~
~~d. ____~~	~~d. ____~~

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

HPSI System - Single Pump

a. Injection Leg 1, greater than or equal to LATER gpm

b. Injection Leg 2, greater than or equal to LATER gpm

c. Injection Leg 3, greater than or equal to LATER gpm

d. Injection Leg 4, greater than or equal to LATER gpm

LPSI System - Single Pump

a. Injection Leg 1, greater than or equal to LATER gpm

b. Injection Leg 2, greater than or equal to LATER gpm

c. Injection Leg 3, greater than or equal to LATER gpm

d. Injection Leg 4, greater than or equal to LATER gpm

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN ~~300~~ 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 high-pressure safety injection pump, and

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a ~~Sump~~ Recirculation Actuation Signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

a. With no ECCS subsystem OPERABLE, restore at least one ECCS subsystem to OPERABLE status within 1 hour or be in COLD SHUTDOWN within the next 20 hours.

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

SURVEILLANCE REQUIREMENTS

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

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

*With pressurizer pressure less than 1700 psia.

#~~A maximum of one high-pressure safety injection pump shall be OPERABLE whenever the temperature of one or more of the RCS cold legs is less than or equal to (275)°F.~~

3/4.5.4 REFUELING WATER STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tanks shall be OPERABLE with:

a. A contained borated water volume of between ~~(464,900)~~ 355,000 and ~~(500,500)~~ (LATER) gallons,

b. Between ~~(1731)~~ 1720 and ~~(2250)~~ 2300 ppm of boron, and

c. A minimum ~~water~~ solution temperature of {35}°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the ~~refueling water tank~~ RWST's inoperable, restore the tanks 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 ~~RWT~~ RWST's shall be demonstrated OPERABLE:

~~a.~~ At least once per 7 days by:

a ~~1.~~ Verifying the contained borated water volume in the tanks, and

b ~~2.~~ Verifying the boron concentration of the water.

~~b. At least once per 24 hours by verifying the RWT temperature when the (outside) air temperature is less than 35°F.~~

c. Verifying the solution temprature.

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SECTION 3/4.~~6F~~

CONTAINMENT SYSTEMS SPECIFICATIONS

~~FOR~~

~~COMBUSTION ENGINEERING~~

~~ATMOSPHERIC TYPE CONTAINMENT~~

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3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

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

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals with gas at P_a (~~54~~ 55.7 psig) and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

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

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.10 percent by weight of the containment air per 24 hours at P_a, (~~54~~ 55.7 psig), or

2. Less than or equal to L_t, 0.05 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, (27.9 psig).

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during shutdown at either P_a (~~54~~ 55.7 psig) or at P_t (~~27~~ 27.9 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet either .75 L_a or .75 L_t, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet either .75 L_a or .75 L_t, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet either .75 L_a or .75 L_t at which time the above test schedule may be resumed.

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

SEE INSERT "A" ATTACHED

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

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

~~3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage at P_a (54 psig) or P_t (27 psig).~~

d. Type B and C tests shall be conducted with gas at P_a (~~54~~ 55.7 psig) at intervals no greater than 24 months except for tests involving:

1. Air locks,

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

2 ~~3~~. Valves pressurized with fluid from a seal system.

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

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

f ~~g~~. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a (~~59.4~~ 61.3 psig) and the seal system capacity is adequate to maintain system pressure for at least 30 days.

INSERT "A"

1. For the superimposed leak test, verifies that the difference between the supplemental and Type A test data is within 0.25 L_a or 0.25 L_t, has a sufficient duration to establish accurately the change in leakage rate between the Type A test and the supplemental test, and requires the quantity of gas bled from the containment during the supplemental test to be equal to at least 25 percent of the total measured leakage at P_a (55.7) psig or P_t (27.9) psig.

2. For the mass step change test, verifies that the metered mass of air bled from or injected into the containment and the change of mass in containment air as measured by the Type A test instrumentation are within 25 percent, does not remove or inject more than 25 percent of the daily allowable leakage in any one hour period, and involves a total metered mass change between 75 and 125 percent of the daily allowable leakage."

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SURVEILLANCE REQUIREMENTS (Continued)

~~h. Type B tests for penetrations employing a continuous leakage monitoring system shall be conducted at P_a (54 psig) at intervals no greater than once per 3 years~~.

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

h ~~j~~. The provisions of Specification 4.0.2 are not applicable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, (~~54~~ 55.7 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one containment air lock door inoperable:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

b. With the containment air lock inoperable, except as the result of an inoperable air lock door, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. (LATER) ~~*After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying no detectable seal leakage by pressure decay when the volume between the door seals is pressurized to greater than or equal to P_a §54 psig§ for at least 15 minutes,~~

b. At least once per 6 months# by conducting an overall air lock leakage test at P_a (~~54~~ 55.7 psig) and by verifying that the overall air lock leakage rate is within its limit, and

c. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

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

#The provisions of Specification 4.0.2 are not applicable.

N/A

DELETE

CONTAINMENT SYSTEMS

CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS (OPTIONAL)

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.4.1 The containment isolation valve pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to 1.10 P_a (59.4 psig) and has adequate capacity to maintain system pressure for at least 30 days.

4.6.1.4.2 The containment channel weld pressurization system shall be demonstrated OPERABLE at least once per 31 days by verifying that the system is pressurized to greater than or equal to P_a (54 psig) and has adequate capacity to maintain system pressure for at least 30 days.

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.~~6~~ 4 Primary containment internal pressure shall be maintained between +1.5 and -0.3 ~~PSIG~~ 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.~~6~~ 4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

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CONTAINMENT SYSTEMS

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.~~6~~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.~~6~~5 The primary containment average air temperature shall be the arithmetical average of the temperatures at any four of the following locations and shall be determined at least once per 24 hours:

Location

a. Elevation 176' - 0"

b. Elevation 68' - 0"

c. Elevation 49' - 6"

d. Elevation 34' - 0"

e. Elevation 19' - 6"

N/A

CONTAINMENT SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

*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.

N/A

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

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

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

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

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

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

N/A

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

CONTAINMENT SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.6.1.~~7~~6 The structural integrity of the containment shall be maintained at a level consistent with the acceptance criteria in Specification 4.6.1.~~7~~6.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.~~7~~6.1 Containment Tendons ~~The~~ Containment tendons' structural integrity shall be demonstrated at the end of one, three and five years following the initial containment structural integrity test and at five year intervals thereafter, following the schedule and format in Table 4.6-1. ~~The~~ Tendons' structural integrity shall be demonstrated by:

a. ~~Determining that a representative sample* of at least 4%, but no less than 4, of the U tendons each have a lift off force of between ____ (minimum) and ____ (maximum) pounds at the first year inspection and that a representative sample* of a least 4%, but no less than 9, of the hoop tendons each have a lift off force of between ____ (minimum) and ____ (maximum) pounds at the first year inspection. For subsequent inspections, the maximum allowable lift off forces shall be decreased from the value determined at the first year inspection by the amount: ____ log t and the minimum allowable lift off force shall be decreased from the value determined at the first year inspection by the amount: ____ log t~~ where t is the time interval in years from initial tensioning of the tendon to the current

[SEE INSERT "A" ATTACHED]

~~*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 observied data, one tendon from each group (U and hoop) may be kept unchanged after the initial selection.~~

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INSERT "A"

a. Determining, by Prestress Monitoring of tendons, that each U-tendon has a lift-off force of between 1393 and 1644 kips and that each hoop tendon has a lift off force between 1281 and 1626 kips at the first year inspection. For subsequent inspections, the maximum allowable lift-off forces shall be decreased from the value determined at the first year inspection by the amount: 22.5 log t (U tendons), 21.8 log t (hoop tendons); the minimum allowable lift-off force shall be decreased from the value determined at the first year inspection by the amount: 28.1 log t (U tendons), 31.8 log t (hoop tendons),

SURVEILLANCE REQUIREMENTS (Continued)

one tendon from each group

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

Performing a Detensioning and Material Test and Inspection by

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

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

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

3. A minimum tensile strength value of 270 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the wire or strand samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

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SURVEILLANCE REQUIREMENTS (Continued)

4.6.1.~~7~~6.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Table 4.6-1 ~~Specification 4.6.1.7.1~~ and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.~~7~~6.3 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces. This inspection shall be performed prior to the Type A containment leakage rate test to verify no apparent changes in appearance or other abnormal degradation.

4.6.1.~~7~~6.4 Reports Any abnormal degradation of the containment structure detected during the above required tests and inspections shall be reported to the Commission pursuant to Specification 6.9.1. This report shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure, the tolerances on cracking, and the corrective actions taken.

TABLE 4.6-1

TENDON SURVEILLANCE

	TENDON NUMBERS									
Years After Initial Structural Integrity Test	1		3		5		10		15	
Type of Inspection	H	U	H	U	H	U	H	U	H	U
Visual Inspection of End Anchorages and Adjacent Concrete Surface	20 86 97 53 64	31-121 9-143 66-176 88-154	5 36 79 113 87	13-139 35-117 4-58 78-164	42 86 75 9 108	64-178 9-143 94-148 19-133	97 86 53	66-176 9-143 39-113	50 114 13	12-140 5-57 96-146
Prestress Monitoring Tests	20 86 97 53 64	31-121 9-143 66-176 88-154			42 86 75 9 108	64-178 9-143 94-148 19-133	97 86 53	66-176 9-143 39-113		
Dentensioning and Material Tests	20	88-154			42	19-133	97	66-176		

	TENDON NUMBERS									
Years After Initial Structural Integrity Test	20		25		30		35		40	
Type of Inspection	H	U	H	U	H	U	H	U	H	U
Visual Inspection of End Anchorages and Adjacent Concrete Surface	75 86 9	86-156 9-143 43-109	12 90 25	24-128 70-172 76-166	86 31 64	9-143 69-178 94-148	81 109 31	41-111 90-152 50-102	20 86 108	9-143 31-121 86-156
Prestress Monitoring Tests	75 86 9	86-156 9-143 43-109			86 31 64	9-143 64-178 94-148			20 86 108	9-143 31-121 86-156
Detensioning and Material Tests	75	43-109			31	64-178			86	9-143

CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY (Reinforced concrete containment)

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

DELETE

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM (OPTIONAL*)

LIMITING CONDITION FOR OPERATION

3.6.1.8 The containment purge supply and exhaust isolation valves shall be closed.

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

ACTION:

With one containment purge supply and/or one exhaust isolation valve open, close the open valve(s) within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SUREVILLANCE REQUIREMENTS

4.6.1.8 The containment purge supply and exhaust isolation valves shall be determined closed at least once per 31 days.

*This specification may be modified if the facility design conforms to Branch Technical Position CSB 6-4 of the Standard Review Plan.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM ~~(Credit taken for iodine removal)~~

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the ~~RWT~~ RWST on a Containment Spray Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is positioned to take suction from the ~~RWT~~ RWST on a Containment ~~Pressure High High test signal.~~ Spray Actuation Signal (CSAS).

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

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

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

2. Verifying that upon a ~~Sump~~ Recirculation Actuation Test Signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump starts automatically on a SAFETY INJECTION ACTUATION test signal.

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

N/A

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM (No credit taken for iodine removal)

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWT on a Containment Spray Actuation Signal and automatically transferring suction to the containment sump on a Sump Recirculation Actuation Signal. Each spray system flow path from the containment sump shall be via an OPERABLE shutdown cooling heat exchanger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

b. With two containment spray systems inoperable and at least (four) containment cooling fans OPERABLE, restore at least one spray system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both spray systems to OPERABLE status within 7 days of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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

N/A

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is positioned to take suction from the RWT on a Containment Pressure--High-High test signal.

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

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

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

2. Verifying that upon a Sump Recirculation Actuation Test Signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

3. Verifying that each spray pump starts automatically on a ____________ test signal.

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

CONTAINMENT SYSTEMS

Iodine Removal system

~~SPRAY ADDITIVE SYSTEM (OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.2.2 The ~~spray additive~~ Iodine Removal system shall be OPERABLE with:

a. A spray additive tank containing a volume of between (LATER) and (LATER) gallons of between 40 and 44 % by weight NaOH solution, with a minimum solution temperature of 58°F and,

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the ~~spray additive~~ iodine removal 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~~ iodine removal system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the NaOH solution temperature.

b ~~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.

c ~~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.

d ~~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 ~~[illegible]~~ Containment Spray Actuation test signal.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

CONTAINMENT SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

2. Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

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

N/A

CONTAINMENT SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each group of containment cooling fans shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

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

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

N/A

3/4.6.3 IODINE CLEANUP SYSTEM (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.6.3 Two independent containment iodine cleanup systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.3 Each iodine cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least (15 minutes) (10 hours with the heaters on).

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

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

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

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

N/A

SURVEILLANCE REQUIREMENTS (Continued)

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

d. At least once per 18 months by:

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

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

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

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

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

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ______ cfm ± 10%.

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

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that on a Containment Radiation-High test signal, all containment purge valves actuate to their isolation position.

4.6.~~4~~3.3 The isolation time of each power operated or automatic valve of Table 3.6-1 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

4.6.~~4~~3.4 Each containment mini purge isolation valve shall be demonstrated OPERABLE within 24 hours after each closing of the valve, except when the valve is being used for multiple cyclings, then at least once per 72 hours, by verifying that when the measured leakage rate is added to the leakage rates determined pursuant to Specification 4.6.1.2.d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

~~SE ATMOSPHERIC~~
SAN ONOFRE - UNIT 2

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME
		( ) seconds
A. CONTAINMENT ISOLATION		
1.		
2.		
B. CONTAINMENT PURGE		
1.		
2.		
C. MANUAL		
1.		
2.		
D. OTHER		
1.		
2.		

*May be opened on an intermittent basis under administrative control.

#Not subject to Type C leakage tests.

SEE INSERTS "A", "B", "C", "D", "E" and "F" ATTACHED

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TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC)
A. CONTAINMENT ISOLATION			
1	HV-0510	Pressurizer steam space sample	10
1	HV-0511	Pressurizer steam space sample	10
2	TV-9267	Letdown line to letdown heat exchanger	5
2	HV-9205	Letdown line to letdown heat exchanger	5
4	HV-0508	Reactor coolant loops hot leg sample	10
4	HV-0509	Reactor coolant loops hot leg sample	10
4	HV-0517	Reactor coolant loops hot leg sample	10
6	HV-9334	Safety injection drain to RWST	10
7	HV-9217	Reactor coolant pump seal bleed off	5
7	HV-9218	Reactor coolant pump seal bleed off	5
11	HV-7911	Demineralized water to service station and sump pump	10
12	HV-0512	Pressurizer surge line sample	10
12	HV-0513	Pressurizer surge line sample	10
13	HV-5803	Containment sump pump discharge	10
13	HV-5804	Containment sump pump discharge	10
14	HV-5686	Fire protection	30
16C	HV-7805	Containment air radioactivity monitor inlet	1
16C	HV-7810	Containment air radioactivity monitor inlet	1
22	HV-5388	Instrument air supply line	10
23A	HV-5437	N_2 supply to quench tank, reactor coolant drain tank, and steam generators	10
26	HV-7512	Reactor coolant drain tank pump discharge	10
26	HV-7513	Reactor coolant drain tank pump discharge	10
27C	HV-7806	Containment air radioactivity monitor outlet	1
27C	HV-7811	Containment air radioactivity monitor outlet	1
27C	HV-7816	Containment air radioactivity monitor outlet	1
28	HV-4052#	Steam generator feedwater	10
29	HV-4048#	Steam generator feedwater	10
30A	HV-7802	Containment air radioactivity monitor inlet	1
30A	HV-7803	Containment air radioactivity monitor inlet	1
30A	HV-7801	Containment air radioactivity monitor outlet	1

INSERT "A"

TABLE 3.6-1
(Continued)
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC)
30b	HV-7800	Containment air radioactivity monitor outlet	1
30c	HV-0516	Quench tank and drain tank gas sample	10
30c	HV-0514	Quench tank and drain tank gas sample	10
30c	HV-0515	Quench tank and drain tank gas sample	10
32	HV-8204#	Mainsteam isolation	5
33	HV-8205#	Mainsteam isolation	5
42	HV-6211	Component cooling water inlet	10
42	HV-6223	Component cooling water inlet	10
43	HV-6216	Component cooling water outlet	10
43	HV-6236	Containment cooling water outlet	10
45	HV-9900	Containment normal A/C chilled water inlet	10
45	HV-9920	Containment normal A/C chilled water inlet	10
46	HV-9971	Containment normal A/C chilled water inlet	10
46	HV-9921	Containment normal A/C chilled water outlet	10
47	HV-7258	Containment waste gas vent header	30
47	HV-7259	Containment waste gas vent header	30
77	HV-5434	Nitrogen supply to safety injection tanks	10
B. CONTAINMENT PURGE			
18	HV-9949	Containment purge inlet (normal)	10
18	HV-9948	Containment purge inlet (normal)	10
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950	Containment purge outlet (normal)	10
19	HV-9951	Containment purge outlet (normal)	10
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

INSERT "B"

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TABLE 3.6-1
(Continued)
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC)
C. MANUAL			
6	2"-099-C-334	Safety injection drain to RWST	NA
8	HV-9200	Charging line to regenerative heat exchanger	NA
9	HV-9337	Shutdown cooling to LPSI pumps	NA
9	HV-9377	Shutdown cooling to LPSI pumps	NA
9	HV-9336	Shutdown cooling to LPSI pumps	NA
9	HV-9379	Shutdown cooling to LPSI pumps	NA
9	PSV-9349	Shutdown cooling to LPSI pumps	NA
10A	HV-0352A#	Containment pressure detectors	NA
10B	3/4"-038-C-396	Integrated leak rate test pressure sensor	NA
10B	3/4"-039-C-396	Integrated leak rate test pressure sensor	NA
16A	HV-0500	Post LOCA hydrogen monitor	NA
16A	HV-0501	Post LOCA hydrogen monitor	NA
16B	HV-0502	Post LOCA hydrogen monitor	NA
16B	HV-0503	Post LOCA hydrogen monitor	NA
20	2"-321-C-376	Quench tank makeup	NA
21	2"-055-C-387	Service air supply line	NA
25	10"-100-C-212	Refueling canal fill and drain	NA
25	10"-101-C-212	Refueling canal fill and drain	NA
27A	HV-0352D#	Containment pressure detectors	NA
31	HV-9946	Containment hydrogen purge inlet	NA
31	HCV-9945	Containment hydrogen purge inlet	NA
40A	HV-0352B#	Containment pressure detectors	NA
42	HV-6223	Component cooling water inlet	NA
43	HV-6236	Component cooling water outlet	NA
67	HV-9434	Hot leg injection	NA
68	2"-130-C-334	Charging line to auxiliary spray	NA
70	2"-037-C-387	Auxiliary steam inlet to utility stations	NA
70	2"-038-C-387	Auxiliary steam inlet to utility stations	NA
71	HV-9420	Hot leg injection	NA
73A	HV-0352C#	Containment pressure detectors	NA
74	HV-9917	Containment hydrogen purge outlet	NA
74	HCV-9918	Containment hydrogen purge outlet	NA

3/4 6-30e

INSERT "C"

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TABLE 3.6-1
(Continued)
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC)
D. OTHER			
3	3"-018-A-551#	High pressure safety injection	NA
3	HV-9323#	High pressure safety injection	NA
3	HV-9324#	High pressure safety injection	NA
5	3"-019-A-551#	High pressure safety injection	NA
5	HV-9326#	High pressure safety injection	NA
5	HV-9327#	High pressure safety injection	NA
8	2"-122-C-554	Charging line to regenerative heat exchanger	NA
11	3"-236-C-675	Demineralized water to service stations and sump pump	NA
14	4"-061-C-681	Fire protection	NA
17	HV-4058#*	Steam generator secondary coolant sample	NA
20	2"-573-C-611	Quench tank makeup	NA
21	2"-017-C-627	Service air supply line	NA
22	1-1/2"-016-C-617	Instrument air supply line	NA
23A	3/4"-002-C-611	LP N_2 to containment	NA
32	HV-8421#	Mainsteam atmospheric dump	NA
32	PSV-8410#	Mainsteam relief	NA
32	PSV-8411#	Mainsteam relief	NA
32	PSV-8412#	Mainsteam relief	NA
32	PSV-8413#	Mainsteam relief	NA
32	PSV-8414#	Mainsteam relief	NA
32	PSV-8415#	Mainsteam relief	NA
32	PSV-8416#	Mainsteam relief	NA
32	PSV-8417#	Mainsteam relief	NA
32	PSV-8418#	Mainsteam relief	NA
32	HV-8249B#	Mainsteam trap isolation	NA
32	HV-8202#	Mainsteam isolation bypass	NA
32	HV-8200#	Mainsteam to auxiliary feedwater turbine	NA
33	HV-8419#	Mainsteam atmospheric dump	NA
33	PSV-8401#	Mainsteam relief	NA
33	PSV-8402#	Mainsteam relief	NA
33	PSV-8403#	Mainsteam relief	NA

INSERT "D"

TABLE 3.6-1
(Continued)
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC)
33	PSV-8404#	Mainsteam relief	NA
33	PSV-8405#	Mainsteam relief	NA
33	PSV-8406#	Mainsteam relief	NA
33	PSV-8407#	Mainsteam relief	NA
33	PSV-8408#	Mainsteam relief	NA
33	PSV-8409#	Mainsteam relief	NA
33	HV-8248B#	Mainsteam trap isolation	NA
33	HV-8203#	Mainsteam isolation bypass	NA
33	HV-8201#	Mainsteam to auxiliary feedwater turbine	NA
36	HV-4054#*	Steam generator blowdown	NA
37	HV-4053#*	Steam generator blowdown	NA
39	3"-020-A-551#	High pressure safety injection	NA
39	HV-9329#	High pressure safety injection	NA
39	HV-9330#	High pressure safety injection	NA
41	3"-021-A-551#	High pressure safety injection	NA
41	HV-9332#	High pressure safety injection	NA
41	HV-9333#	High pressure safety injection	NA
44	HV-4057#*	Steam generator secondary coolant sample	NA
48	8"-072-A-552#	Low pressure safety injection	NA
48	HV-9322#	Low pressure safety injection	NA
49	8"-073-A-552#	Low pressure safety injection	NA
49	HV-9325#	Low pressure safety injection	NA
50	8"-074-A-552#	Low pressure safety injection	NA
50	HV-9328#	Low pressure safety injection	NA
51	8"-075-A-552#	Low pressure safety injection	NA
51	HV-9331#	Low pressure safety injection	NA
52	8"-004-C-406	Containment spray inlet	NA
52	HV-9367	Containment spray inlet	NA
53	8"-006-C-406	Containment spray inlet	NA
53	HV-9368	Containment spray inlet	NA
54	HV-9304	Containment emergency sump recirculation	NA
54	HV-9302	Containment emergency sump recirculation	NA

INSERT "E"

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TABLE 3.6-1
(Continued)
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	ISOLATION TIME (SEC)
55	HV-9305	Containment emergency sump recirculation	NA
55	HV-9303	Containment emergency sump recirculation	NA
56	HV-6366	Containment emergency A/C cooling water inlet	NA
57	HV-6372	Containment emergency A/C cooling water inlet	NA
58	HV-6368	Containment emergency A/C cooling water inlet	NA
59	HV-6370	Containment emergency A/C cooling water inlet	NA
60	HV-6369	Containment emergency A/C cooling water outlet	NA
61	HV-6371	Containment emergency A/C cooling water outlet	NA
62	HV-6367	Containment emergency A/C cooling water outlet	NA
63	HV-6373	Containment emergency A/C cooling water outlet	NA
67	3"-157-A-551	Hot leg injection	NA
68	2"-129-A-554	Charging line to auxiliary spray	NA
71	3"-158-A-551	Hot leg injection	NA
75	HV-4715#	Steam generator auxiliary feedwater	NA
75	HV-4731#	Steam generator auxiliary feedwater	NA
77	2"-108-C-627	Nitrogen supply to safety injection tanks	NA
78	HV-4714	Steam generator auxiliary feedwater	NA
78	HV-4730	Steam generator auxiliary feedwater	NA

INSERT "F"

*May be opened on an intermittent basis under administrative control.

#Not subject to Type C leakage tests.

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CONTAINMENT SYSTEMS

3/4.6.~~5~~4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.~~5~~4.1 Two independent containment hydrogen analyzers shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.~~5~~4.1 Each hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS - W

LIMITING CONDITION FOR OPERATION

3.6.~~5~~4.2 Two independent containment hydrogen recombiner systems shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.~~5~~4.2 Each hydrogen recombiner system shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying, during a recombiner system functional test, that the minimum heater sheath temperature increases to greater than or equal to 700°F within 90 minutes ~~. 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.~~ and is maintained for atleast 2 hours.

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 recombiners ~~enclosure~~ (i.e., loose wiring or structural connections, deposits of foreign materials, etc.), and

3. Verifying the integrity of the heater electrical circuits by performing a continuity and 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.

N/A

CONTAINMENT SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.5.3 The hydrogen purge cleanup system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes (10 hours with the heaters on).

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

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

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

N/A

SURVEILLANCE REQUIREMENTS (Continued)

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

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

d. At least once per 18 months by:

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

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

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

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

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ____ cfm ± 10%.

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

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CONTAINMENT DOME AIR CIRCULATORS ~~HYDROGEN MIXING SYSTEM (OPTIONAL)~~

LIMITING CONDITION FOR OPERATION

3.6.~~5.4~~ 4.3 Two independent ~~hydrogen mixing systems~~ dome air circulator trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one ~~hydrogen mixing system~~ dome air circulator train inoperable, restore the inoperable ~~system~~ train to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.~~5.4~~ 4.3 Each ~~hydrogen mixing system~~ dome air circulating train shall be demonstrated OPERABLE:

a. At least once per ~~92 days on a STAGGERED TEST BASIS~~ 18 months by starting each ~~system~~ train ~~from the control room~~ on a CCAS signal and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by verifying a system flow rate of at least 37,000 cfm.

N/A

CONTAINMENT SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes (10 hours with the heaters on).

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

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

   2. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ____ cfm ± 10%.

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N/A

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

d. At least once per 18 months by:

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

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

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

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

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

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of ____ cfm ± 10%.

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

N/A

CONTAINMENT SYSTEMS

3/4.6.7 VACUUM RELIEF VALVES (OPTIONAL)

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With both 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 Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

~~b. With one reactor coolant loop and associated steam generator in operation and with one or more main steam line code safety valves associated with the operating steam generator inoperable, operation in MODES 1, 2 and 3 may proceed provided:~~

~~1. That at least (2) main steam line code safety valves on the nonoperating steam generator are OPERABLE, and~~

~~2. That within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-3; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.~~

b ~~c~~. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER Line No. 1	VALVE NUMBER Line No. 2	LIFT SETTING (± 1%)*	ORIFICE SIZE
a.	2PSV-8401	2PSV-8410	1100 psia	16 in²
b.	2PSV-8402	2PSV-8411	1107 psia	16 in²
c.	2PSV-8403	2PSV-8412	1114 psia	16 in²
d.	2PSV-8404	2PSV-8413	1121 psia	16 in²
e.	2PSV-8405	2PSV-8414	1128 psia	16 in²
f.	2PSV-8406	2PSV-8415	1135 psia	16 in²
g.	2PSV-8407	2PSV-8416	1142 psia	16 in²
h.	2PSV-8408	2PSV-8417	1149 psia	16 in²
i.	2PSV-8409	2PSV-8418	1155 psia	16 in²

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

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TABLE 3.7-2

MAXIMUM ALLOWABLE LINIAR POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Liniar Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	~~(99.5)~~ 90
2	~~(74)~~ 79
3	~~(48.6)~~ 68
4	57
5	45
6	34
7	23
8	0

N/A

TABLE 3.7-3

MAXIMUM ALLOWABLE LINIAR POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH ONE STEAM GENERATOR

Maximum Number of Inoperable Safety Valves on The Operating Steam Generator	Maximum Allowable Liniar Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	(49)
2	(42)
3	(35)

PLANT SYSTEMS

AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

SEE INSERT "A" ATTACHED

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

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

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

INSERT

~~With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two capable of being powered from separate emergency busses and one capable of being powered by an OPERABLE steam supply system) 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.~~

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. By testing pursuant to Specification 4.0.5.

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 (LATER) psig at a flow of greater than or equal to (LATER) gpm.

2. Verifying that the turbine driven pump develops a discharge pressure of greater than or equal to (LATER) psig at a flow of greater than or equal to (LATER) gpm when the secondary steam supply pressure is greater than (LATER) psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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

INSERT "A"

3.7.1.2 Two Auxiliary feedwater pumps and associated, flow paths shall be OPERABLE with:

a. One motor driven pump capable of being powered from an OPERABLE emergency bus, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one Auxiliary feedwater pump inoperable, restore the inoperable pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the next 12 hours.

~~INSERT~~ DELETE

DELETE

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

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

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

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

2. Verifying that each pump starts automatically upon receipt of an EFAS test signal.

PLANT SYSTEMS

CONDENSATE STORAGE TANKS (CSTs)

LIMITING CONDITION FOR OPERATION

(LATER) (CSTs)

3.7.1.3 The condensate storage tanks ~~(CST)~~ shall be OPERABLE with a contained volume of at least ~~(160,000)~~ gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tanks inoperable, within 4 hours either:

a. Restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours. ~~or~~

~~b. Demonstrate the OPERABILITY of the (alternate water source) as a backup supply to the auxiliary feedwater pumps and restore the condensate storage tank to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.~~

SURVEILLANCE REQUIREMENTS

4.7.1.3.1 The condensate storage tanks shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits. ~~when the tank is the supply source for the auxiliary feedwater pumps.~~

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

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microcuries/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

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PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subseqent operation in MODES 1, 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 at least HOT STANDBY with 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 ~~(3.0)~~ 5.0 seconds when tested pursuant to Specification 4.0.5.

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PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the primary and secondary coolants in the steam generators shall be greater than ~~(90)~~ 70°°F when the pressure of either coolant in the steam generator is greater than ~~(275)~~ 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 ~~(275)~~ 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 generators shall be determined to be less than ~~275~~ 200 psig at least once per hour when the temperature of either the primary or secondary coolant is less than ~~90°~~ 70°F.

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PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on an SIAS test signal.

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PLANT SYSTEMS

3/4.7.4 SALT WATER COOLING SYSTEM ~~SERVICE WATER SYSTEM~~

LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent salt water cooling loops ~~service water loops~~ shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one salt ~~service~~ water cooling 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 salt ~~service~~ water cooling loops shall be demonstrated OPERABLE:

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

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety related equipment actuates to its correct position on a SIAS test signal.

N/A

PLANT SYSTEMS

3/4.7.5 ULTIMATE HEAT SINK (OPTIONAL)

LIMITING CONDITION FOR OPERATION

3.7.5 The ultimate heat sink shall be OPERABLE with:

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

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

N/A

PLANTS SYSTEMS

3/4.7.6 FLOOD PROTECTION (OPTIONAL*)

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

b. Initiate and complete within ______ hours, the following flood protection measures:

1. (Plant dependent)
2. (Plant dependent)

SURVEILLANCE REQUIREMENTS

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

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

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

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

3/4.7.~~1~~ 5 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~1~~ 5 Two independent control room emergency air cleanup systems shall be OPERABLE.

APPLICABILITY: ALL MODES

ACTION:

MODES 1, 2, 3 and 4

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

MODES 5 and 6

a. With one control room emergency air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the control room emergency air cleanup system in the recirculation mode.

b. With both control room emergency air cleanup systems inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

c. The provisions of Specification 3.0.3 are not applicable in MODE 6.

SURVEILLANCE REQUIREMENTS

4.7.~~1~~ 5 Each control room emergency air cleanup system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to ~~(120)~~ 110 °F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least ~~15 minutes (~~10 hours with the heaters on~~)~~.

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

   1. Verifying that with the system operating at a flow rate of 35485 cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake. ~~(For systems with diverting valves.)~~

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 1000 cfm ± 10%, ventilation unit, and 35,485 cfm ± 10% air conditioning unit.

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

4. Verifying a system flow rate of 1000 cfm ± 10% ventilation unit, and 35485 cfm ± 10% air conditioning unit during system operation when tested in accordance with ANSI N510-1975.

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

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ± 10%, ventilation unit, and 35485 cfm air conditioning unit.

2. Verifying that on a ~~containment~~ control room isolation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to ~~(1/4)~~ (LATER) inch W.G. relative to the outside atmosphere during system operation.

4. Verifying that the heaters dissipate 3.2 kw ± 5% ~~kw~~ when tested in accordance with ANSI N510-1975.

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SURVEILLANCE REQUIREMENTS (Continued)

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]%*~~ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 1000 cfm ± 10%, ventilation unit, and 35,485 cfm ±10 % air conditioning unit.

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 1000 cfm ± 10%, ventilation unit, and 35,485 cfm ± 10%, air conditioning unit.

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

N/A

PLANT SYSTEMS

3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes (10 hours with the heaters on).

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

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

N/A

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is ____ cfm ± 10%.

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

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

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

d. At least once per 18 months by:

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

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

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

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

N/A

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

f. After each complete or partial replacement of a charcoal absorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of _____ cfm ± 10%.

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

3/4.7.~~8~~ 6 SNUBBERS

[LATER]

LIMITING CONDITION FOR OPERATION

3.7.~~8~~ 6 All snubbers listed in Tables 3.7-4a and 3.7-4b shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4. (MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES).

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9.c on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

4.7.~~8~~ 6 Each snubber shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. Visual Inspections

The first inservice visual inspection of snubbers shall be during the first COLD SHUTDOWN exceeding 24 hours after four months of power operation and shall include all snubbers listed in Tables 3.7-4a and 3.7-4b. If less than two (2) snubbers are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The snubbers may be categorized into two groups: Those accessible and those inaccessible during reactor operation. Each group may be inspected independently in accordance with the above schedule.

*The inspection interval shall not be lengthened more than one step at a time.

#The provisions of Specification 4.0.2 are not applicable.

b. Visual Inspection Acceptance Criteria [LATER]

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure, and (3) the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, providing that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specifications 4.7.~~9~~6.d or 4.7.~~9~~6.e, as applicable. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as inoperable snubbers.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (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 ~~number of~~ snubber~~s~~ which does ~~do~~ not meet the functional test acceptance criteria of Specification 4.7.~~9~~6.d or 4.7.~~9~~6.e, an additional 10% of that type of snubber shall be functionally tested).

or

(that number of snubbers which follows the expression 35 $(1 + \frac{c}{2})$, where c* is the allowable number of snubbers not meeting the

*The value c will be arbitrarily chosen by the applicant and incorporated into the expressions for the representative sample and for the resample prior to the issuance of the Technical Specifications. The expressions are intended for use in plants with larger numbers of safety-related snubbers (>500) and provide a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptable limits. That is, the confidence level will be provided no matter what value is chosen for c. It is advised, however, that discretion be used when initially choosing the value for c because the lower the value of c (the lower the amount of snubbers in the representative sample), the higher the amount of snubbers required in the re-sample will be. To illustrate: If c = 2 and 3 snubbers are found not to meet the functional test acceptance criteria, there will be 70 snubbers in the representative sample and 31 snubbers required for testing in the re-sample; If c = 2 and 4 snubbers fail the functional test, there will be 70 snubbers in the representative sample and 62 snubbers required for testing in the re-sample; If c = 0 and 1 snubber fails the functional test, there will be 35 snubbers in the representative sample and 140 snubbers required for testing in the re-sample; If c = 0 and 2 snubbers fail the functions test, there will be 35 snubbers in the representative sample and 280 snubbers required for testing in the re-sample.

acceptance criteria selected by the operator, shall be functionally tested either in-place or in a bench test. For each number of snubbers above c which does not meet the functional test acceptance criteria of Specifications 4.7.9.d. or 4.7.9.e, an additional sample selected according to the expression $35 (1 + \frac{c}{2}) (\frac{2}{c + 1})^2 (a - c)$ shall be functionally tested, where a is the total number of snubbers found inoperable during the functional testing of the representative sample.

Functional testing shall continue according to the expression $b [35 (1 + \frac{c}{2}) (\frac{2}{c+1})^2]$ where b is the number of snubbers found inoperable in the previous re-sample, until no additional inoperable snubbers are found within a sample or until all snubbers in Table 3.7-4a and 3.7-4b have been functionally tested).

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. As part of this initial sample, at least 25% of the snubbers in each of the following three categories shall be included:

1. The first snubber away from each reactor vessel nozzle
2. Each snubber within five feet of heavy equipment (valve, pump, turbine, motor, etc.)
3. Each snubber within ten feet of the discharge from a safety relief valve

Snubbers identified in Tables 3.7-4a and 3.7-4b as "Especially Difficult to Remove" or in "High Radiation Zones During Shutdown" shall also be included in the representative sample.* Tables 3.7-4a and 3.7-4b may be used jointly or separately as the basis for the sampling plan.

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the re-sampling.

*Permanent or other exemptions from functional testing for individual snubbers in these categories may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

(LATER)

SURVEILLANCE REQUIREMENTS (Continued)

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, all snubbers of the same design shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are supported by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components supported by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the supported component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force. The differential of this force shall not exceed 50% between two consecutive tests.

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

SURVEILLANCE REQUIREMENTS (Continued) (LATER)

f. Snubber Service Life Monitoring

A record of the service life of each snubber, the date at which the designated service life commences and the installation and maintenance records on which the designated service life is based shall be maintained as required by Specification 6.10.2.1.

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and maintenance records for each snubber listed in Tables 3.7-4a and 3.7-4b shall be reviewed to verify that the indicated service life has not been exceeded or will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded prior to the next scheduled snubber service life review, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned so as to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement or reconditioning shall be indicated in the records.

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TABLE 3.7-4a (LATER)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE DURING SHUTDOWN** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-4a provided that a revision to Table 3.7-4a is included with the next License Amendment request.

**Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-4a is included with the next License Amendment request.

TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE DURING SHUTDOWN** (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)

* Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-4b provided that a revision to Table 3.7-4b is included with the next License Amendment request.

**Modifications to this column due to changes in high radiation areas may be made without prior License Amendment provided that a revision to Table 3.7-4b is included with the next License Amendment request.

3/4.7.~~10~~7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.~~10~~7 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of greater than or equal to 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

    1. Either decontaminate and repair the sealed source, or

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

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

SURVEILLANCE REQUIREMENTS

4.7.~~10~~7.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

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

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

4.7.~~10~~7.2 Test Frequencies - Each category of sealed sources (excluding startup sources and fission detectors previously subjected to core flux) shall be tested at the frequencies described below.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

2. In any form other than gas.

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.

4.7.~~10~~7.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of greater than or equal to 0.005 microcuries of removable contamination.

PLANT SYSTEMS

3/4.7.~~11~~ 8 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~11~~ 8.1 The fire suppression water system shall be OPERABLE with:

a. {Two} electric motor-driven fire ~~suppression~~ pumps, of 1500 gal/min capacity or one desel-driven fire pump of ~~each with a capacity of~~ {2500} gpm CAPACITY, with their discharge aligned to the fire suppression header,

b. Two Separate water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An OPERABLE flow path capable of taking suction from the water supply ~~tank~~ per 3.7.8.1 ~~and the _____ tank~~ and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the first ~~last~~ valve upstream ~~ahead~~ of the water flow alarm device on each sprinkler, ~~or~~ hose standpipe, ~~and the last valve ahead of the deluge valve on each deluge~~ or spray system riser required to be OPERABLE per Specification ~~s~~ ~~3.7.11.2, 3.7.11.5 and 3.7.11.6~~. 3.7.8.2

APPLICABILITY: At all times.

ACTION:

a. With one required electric motor-driven pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

1. Establish a backup fire suppression water system within 24 hours, and

2. In lieu of any other report required by Specification 6.9.1, submit a Special Report in accordance with Specification 6.9.2:

   a) By telephone within 24 hours,

   b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

PLANT SYSTEMS

3/4.7.~~11~~ 8 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~11~~ 8.1 The fire suppression water system shall be OPERABLE with:

a. {Two} electric motor-driven fire ~~suppression~~ pumps of 1500 gal/min capacity or one diesel-driven fire pump of ~~each with a capacity of~~ {2500} gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An OPERABLE flow path capable of taking suction from the water ~~tank~~ supply per 3.7.8.1.1 ~~and the _____ tank~~ and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the ~~last~~ first valve upstream ~~ahead~~ of the water flow alarm device on each sprinkler, ~~or~~ hose standpipe, ~~and the last valve ahead of the deluge valve on each deluge~~ or spray system(s) required to be OPERABLE per Specifications ~~3.7.11.2, 3.7.11.5 and 3.7.11.6~~. 3.7.8.2

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable:

1. Establish a backup fire suppression water system within 24 hours, and

2. In lieu of any other report required by Specification 6.9.1, submit a Special Report in accordance with Specification 6.9.2:

   a) By telephone within 24 hours,

   b) Confirmed by telegraph, mailgram or facsimile transmission no later than the first working day following the event, and

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ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

4.7.~~11~~8.1.1 The fire suppression water system shall be demonstrated OPERABLE:

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

b. At least once per ~~31~~ 92 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

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

d. {At least once per 6 months by performance of a system flush.}

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

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

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

1. ~~2.~~ Verifying that each pump develops at least ~~(2500) gpm at a system head of (250) feet,~~ 93% of its flow and head at some point on the manufacturers performance curves,

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

3. ~~4.~~ Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 95 psig.

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SURVEILLANCE REQUIREMENTS (Continued)

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

4.7.~~11~~8.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per ~~31~~ 92 days by verifying:

1. The diesel fuel oil day storage tank contains at least 225 gallons of fuel, and

2. The diesel starts from ambient conditions and operates for at least 30 minutes on recirculation flow.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank, obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

c. At least once per 18 months during shutdown, by subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for the class of service.

4.7.~~11~~8.1.3 The fire pump diesel starting 24-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each battery is above the plates, and

2. The overall battery voltage is greater than or equal to 24 volts.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days by verifying that the specific gravity is appropriate for continued service of the battery.

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

1. The batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioriation, and

2. The battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.~~11~~ 9.2 The following spray and/or sprinkler systems shall be OPERABLE:

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

~~b.~~ SEE INSERT "A" ATTACHED.

~~c.~~

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.7.~~11~~ 9.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

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

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

Insert "A"

Hazard	Location	No. of Systems	System Type
Reactor Coolant Pumps	Containment	4	Deluge-Water Spray
R.R. Tunnel	Fuel Hand. Bldg.	1	Wet Pipe
Truck Ramp	Radwaste Bldg.	1	Wet Pipe
Cable Tunnel	Section 1	1	Deluge-Water Spray
Cable Tunnel	Section 2	1	Deluge-Water Spray
Cable Tunnel	Section 3	1	Deluge-Water Spray
Cable Tunnel	Section 4	1	Deluge-Water Spray
Cable Tunnel	Section 5	1	Deluge-Water Spray
Cable Tunnel	Section 6	1	Deluge-Water Spray
Cable Tunnel	Section 7	1	Deluge-Water Spray
Cable Tunnel	Section 8	1	Deluge-Water Spray
Cable Tunnel	Section 9	1	Deluge-Water Spray
Cable Tunnel	Section 10	1	Deluge-Water Spray
Cable Tunnel Riser	Fuel Hand Bldg.	1	Deluge-Water Spray
Cable Gallery	Radwaste Bldg.	2	Deluge-Water Spray
Cable Risers El. 9 ft.	Control Bldg.	2	Deluge-Water Spray
Cable Risers El. 30 ft.	Control Bldg.	2	Deluge-Water Spray
Cable Risers El. 50 ft.	Control Bldg.	2	Deluge-Water Spray
Cable Risers El. 70 ft.	Control Bldg.	2	Deluge-Water Spray
Cable Spreading Room	Control Bldg.	4	Deluge-Water Spray

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months:

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

   a) Verifying that the automatic valves in the flow path actuate to their correct positions on a ~~[illegible]~~ test signal, and

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

2. By a visual inspection of the dry pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each nozzle's spray area to verify the spray pattern is not obstructed.

d. At least once per 3 years by performing an air flow test through each open head spray/sprinkler header and verifying each open head spray/sprinkler nozzle is unobstructed.

PLANT SYSTEMS

CO_2 SYSTEMS

LIMITING CONDITION FOR OPERATION

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

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

b.

c.

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.7.11.3.1 Each of the above required CO_2 systems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path is in its correct position.

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

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

N/A

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 18 months by verifying:

1. The system valves and associated ventilation dampers and fire door release mechanisms actuate manually and automatically, upon receipt of a simulated actuation signal, and

2. Flow from each nozzle during a "Puff Test."

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

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

b. At least once per 18 months by:

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

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

N/A

PLANT SYSTEMS

HALON SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.4 The following Halon systems shall be OPERABLE with the storage tanks having at least 95% of full charge weight (or level) and 90% of full charge pressure.

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

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

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 6 months by verifying Halon storage tank weight (level) and pressure.

c. At least once per 18 months by:

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

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

PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.~~11~~8.5 The fire hose stations shown in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ACTION:

a. With one or more of the fire hose stations OUTSIDE CONTAINMENT shown in Table 3.7-5 inoperable, route an additional equivalent capacity fire hose to the unprotected area(s) from an OPERABLE hose station within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise route the additional hose within 24 hours. Restore the fire hose station to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the station to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.~~11~~8.5 Each of the fire hose stations shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations OUTSIDE of CONTAINMENT THAT are accessible during plant operation to assure all required equipment is at the station.

b. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.
   2. Removing the hose for inspection and re-racking, and
   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

c. At least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.
   2. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any hose station.

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TABLE 3.7-5

FIRE HOSE STATIONS

LOCATION*	ELEVATION	HOSE RACK IDENTIFICATION

SEE INSERT "A" ATTACHED

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

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INSERT "A"

TABLE 3.7-5

FIRE HOSE STATIONS

LOCATION	ELEVATION	STATION NUMBER
Containment Bldg. - Unit 2	63'-6"	130
Containment Bldg. - Unit 2	63'-6"	1
Containment Bldg. - Unit 2	63'-6"	8
Containment Bldg. - Unit 2	45'-0"	2
Containment Bldg. - Unit 2	45'-0"	5
Containment Bldg. - Unit 2	45'-0"	9
Containment Bldg. - Unit 2	30'-0"	3
Containment Bldg. - Unit 2	30'-0"	6
Containment Bldg. - Unit 2	30'-0"	10
Containment Bldg. - Unit 2	17'-6"	4
Containment Bldg. - Unit 2	17'-6"	7
Containment Bldg. - Unit 2	17'-6"	11
Electrical Penetration Area - Unit 2	45'-0"	120
Electrical Penetration Area - Unit 2	45'-0"	121
Electrical Penetration Area - Unit 2	63'-6"	122
Electrical Penetration Area - Unit 2	63'-6"	123
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	9'-0"	109
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	9'-0"	114
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	108
Cable Spreading Room-Auxiliary Bldg. Control Area	9'-0"	113
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	48
Cable Spreading Room Corridor-Auxiliary Bldg. Control Area	9'-0"	60
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	30'-0"	110
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	30'-0"	115
Corridor (North)-Auxiliary Bldg. Control Area	30'-0"	49
Corridor (South)-Auxiliary Bldg. Control Area	30'-0"	61
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	50'-0"	111
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	50'-0"	116
Corridor (North)-Auxiliary Bldg. Control Area	50'-0"	50
Corridor (South)-Auxiliary Bldg. Control Area	50'-0"	62
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	56
HVAC Room Corridor-Auxiliary Bldg. Control Area	50'-0"	57
Cable Riser Gallery (North)-Auxiliary Bldg. Control Area	70'-0"	112
Cable Riser Gallery (South)-Auxiliary Bldg. Control Area	70'-0"	117
Fuel Handling Bldg.-Unit 2	63'-6"	118
Fuel Handling Bldg.-Unit 2	63'-6"	119

N/A

PLANT SYSTEMS

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.6 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenver equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-6 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise provide the additional hose within 24 hours. Restore the hydrant or hose house to OPERABLE status within 14 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the hydrant or hose house to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.7.11.6 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-6 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months (once during March, April or May and once during September, October or November) by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

1. Conducting a hose hydrostatic test at a pressure at least 50 psig greater than the maximum pressure available at any yard fire hydrant.

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

3. Performing a flow check of each hydrant to verify its OPERABILITY.

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N/A

~~TABLE 3.7-6~~

~~YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES~~

~~LOCATION*~~	~~HYDRANT NUMBER~~

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

3/4.7.~~12~~9 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.~~12~~9 All fire barrier penetrations (including cable penetration barriers, fire doors and fire dampers), in fire zone boundaries, protecting safety related areas shall be functional.

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire barrier penetrations non-functional, within one hour either, establish a continuous fire watch on at least one side of the affected penetration or verify the OPERABILITY of fire detectors on at least one side of the non-functional fire barrier and establish an hourly fire watch patrol. Restore the non-functional fire barrier penetration(s) to functional status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the action taken, the cause of the non-functional penetration and plans and schedule for restoring the fire barrier penetration(s) to functional status.

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

SURVEILLANCE REQUIREMENTS

4.7.~~12~~9 Each of the above required fire barrier penetrations shall be verified to be functional:

a. At least once per 18 months by a visual inspection.

b. Prior to returning a fire barrier penetration to functional status following repairs or maintenance by performance of a visual inspection of the affected fire barrier penetration(s).

N/A

PLANT SYSTEMS

3/4.7.13 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.13 The temperature of each area shown in Table ~~3.77~~ 3.7-7 shall be maintained within the limits indicated in Table 3.7-7.

APPLICABILITY: Whenever the equipment in an affected area is required to be OPERABLE.

ACTION:

With one or more areas exceeding the temperature limit(s) for equipment not operating shown in Table 3.7-7 for more than 4 hours:

a. Declare the equipment in the area inoperable and apply the appropriate ACTION requirement(s) for the inoperable equipment, and

b. Perform an engineering evaluation to determine the effects of the out of limit temperature on the service life of the equipment located in the area.

SURVEILLANCE REQUIREMENTS

4.7.13 The temperature in each of the areas of Specification 3.7.13 shall be determined to be within its limit at least once per 24 hours.

N/A

~~TABLE 3.7-7~~

~~AREA TEMPERATURE MONITORING~~

~~AREA~~	~~TEMPERATURE LIMIT (°F)~~
~~1.~~	
~~2.~~	
~~3.~~	
~~4.~~	
~~5.~~	

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators each with:

1. A ~~Separate~~ day ~~and engine-mounted~~ fuel tank~~s~~ containing a minimum volume of 325 gallons of fuel,

2. A separate fuel storage system containing a minimum volume of 48,736 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With either an offsite circuit or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ELECTRICAL POWER SYSTEMS

ACTION (Continued)

c. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2.a.4 within one hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availablity, and

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring {manually and automatically} unit power supply from the normal circuit to the alternate circuit.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

a. In accordance with the frequency specified in Table 4.8-1 on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the day ~~and engine-mounted~~ fuel tank,

2. Verifying the fuel level in the fuel storage tank,

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day ~~and engine-mounted~~ tank,

4. Verifying the diesel ~~starts~~ can be manually started from ambient condition and will accelerate~~s~~ to at least {900} rpm in less than or equal to {10} seconds. The generator voltage and frequency shall be 4360 ~~(4160)~~ ± 436 ~~(420)~~ volts and {60} ± {1.2} Hz within {10} seconds after the start signal. ~~The diesel generator shall be started for this test by using one of the following signals with startup on each signal verified at least once per 124 days.~~

   ~~a) Manual.~~

   ~~b) Simulated loss of offsite power by itself.~~

   ~~c) Simulated loss of offsite power in conjunction with an ESF actuation test signal.~~

   ~~d) An ESF actuation test signal by itself.~~

5. Verifying the generator is synchronized, loaded to greater than or equal to 4700 ~~(continuous rating)~~ kw in less than or equal to 77 ~~(60)~~ seconds, and operates for greater than or equal to 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 92 days by verifying that a sample of diesel fuel from the fuel storage tank obtained in accordance with ASTM-D270-65, is within the acceptable limits specified in Table 1 of ASTM D975-74 when checked for viscosity, water and sediment.

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

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a load of greater than or equal to (~~largest single emergency load~~) (655.7) kw for Generator G002 and (478) kw for Generator G003 kw while maintaining voltage at (~~4160~~) 4360 ± (~~420~~) 436 volts and frequency at (60) ± (~~1.2~~) 5.25 Hz (less than or equal to 75% of the difference between nominal speed and the overspeed trip setpoint, or 15% above nominal, whichever is less).

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ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the generator capability to reject a load of 4700 ~~(continuous rating)~~ kw without tripping. ~~The generator voltage shall not exceed _____ volts during and following the load rejection.~~

4. Simulating a loss of offsite power by itself, and:

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds, ~~energizes the auto-connected shutdown loads through the load sequencer~~ and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4360 ~~(4160)~~ ± 500 ~~(420)~~ volts and 60 ± 3.5 ~~(1.2)~~ Hz during this test.

5. Verifying that on an SIAS ~~ESF actuation~~ test signal (without loss of offsite power) the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady state generator voltage and frequency shall be 4360 ± 436 ~~(4160) ± (420)~~ volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and freqency shall be maintained within these limits during this test.

6. Verifying that on a simulated loss of the diesel generator (with offsite power not available), the loads are shed from the emergency busses and that subsequent loading of the diesel generator is in accordance with design requirements.

7. Simulating a loss of offsite power in conjunction with an ESF actuation test signal, and

   a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads within 10 seconds,

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SURVEILLANCE REQUIREMENTS (Continued)

energizes the auto connected emergency ~~(accident)~~ loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at ~~(4160)~~ 4360 ± ~~(420)~~ 1090 volts and {60} ± ~~(1.2)~~ 3 Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to ~~(2-hour rating)~~ 5170 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to ~~(continuous rating)~~ 4700 kw. Within 5 minutes after completing this 24 hour test, perform Specification 4.8.1.1.2.c.4. The generator voltage and freqency shall be ~~(4160)~~ 4360 ± ~~(420)~~ 1090 volts and {60} ± ~~(1.2)~~ 3 Hz within {10} seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

~~9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of ___ kw.~~

9 ~~10~~. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

10 ~~11~~. Verifying that with the diesel generator operating in a test mode {connected to its bus}, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11 ~~12~~. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day ~~and engine-mounted~~ tank of each diesel via the installed cross connection lines.

energizes the auto connected emergency ~~(accident)~~ loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at ~~(4160)~~ 4360 ± ~~(420)~~ 1090 volts and {60} ± ~~(1.2)~~ 3 Hz during this test.

c) Verifying that all automatic diesel generator trips, except engine overspeed and generator differential, are automatically bypassed upon loss of voltage on the emergency bus concurrent with a safety injection actuation signal.

8. Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to greater than or equal to ~~(2-hour rating)~~ 5170 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to ~~(continuous rating)~~ 4700 kw. Within 5 minutes after completing this 24 hour test, perform Specification 4.8.1.1.2.c.4. The generator voltage and freqency shall be ~~(4160)~~ 4360 ± ~~(420)~~ 1090 volts and {60} ± ~~(1.2)~~ 3 Hz within {10} seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

9. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of ___ kw. (LATER)

10. Verifying the diesel generator's capability to:

a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

b) Transfer its loads to the offsite power source, and

c) Be restored to its standby status.

11. Verifying that with the diesel generator operating in a test mode {connected to its bus}, a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

12. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the day ~~and engine-mounted~~ tank of each diesel via the installed cross connection lines.

SURVEILLANCE REQUIREMENTS (Continued)

13. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval.

~~14. Verifying that the following diesel generator lockout features prevent diesel generator starting only when required:~~

~~a) (Later)~~
~~b) (Later)~~

d. 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 {900} rpm in less than or equal to {10} seconds.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

14. Verifying that the following non-critical diesel-generator lockout features prevent diesel start when actuated by a LOVS or manual start:

a) Jacket coolant high-temperatures
b) Volts per cycle high.
c) Tripping relay
d) High Crankcase pressure
e) Diesel generator motoring
f) Generator ground overcurrent
g) Generator voltage restrained overcurrent

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TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last 100 Valid Tests.*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last 100 valid tests". Entry into this test schedule shall be made at the 31 day test frequency.

ELECTRICAL POWER SYSTEMS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day ~~and engine mounted~~ fuel tanks containing a minimum volume of 325 gallons of fuel,

2. A fuel storage system containing a minimum volume of 48736 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for requirement 4.8.1.1.2.a.5) and 4.8.1.1.3.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized: ~~with tie breakers open between redundant busses:~~

(4160)	volt Emergency Bus #	2A04
(4160)	volt Emergency Bus #	2A06
(480)	volt Emergency Bus #	2B04
(480)	volt Emergency Bus #	2B06
(120)	volt A.C. Vital Bus #	2YV1
(120)	volt A.C. Vital Bus #	2YV2
(120)	volt A.C. Vital Bus #	2YV3
(120)	volt A.C. Vital Bus #	2YV4

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE ~~with tie breakers open between redundant busses~~ at least once per 7 days by verifying correct breaker alignment and indicated power availability.

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE:

1 - {4160} volt Emergency Bus

1 - {480} volt Emergency Bus

2 - {120} volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6

ACTION:

With less than the above complement of A.C. busses OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

[SEE INSERT "A" ATTACHED]

ELECTRICAL POWER SYSTEMS

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be energized and OPERABLE:

TRAIN "A" consisting of (250/125)-volt D.C. bus No. 1, (250/125)-volt D.C. battery bank No. 1 and a full capacity charger.

TRAIN "B" consisting of (250/125)-volt D.C. bus No. 2, (250/125)-volt D.C. battery bank No. 2 and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one (250/125)-volt D.C. bus train inoperable, restore the inoperable bus train to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized with tie breakers open at least once per 7 days by verifying correct breaker alignment and indicated power availability with an overall voltage of greater than or equal to (250/125) volts.

4.8.2.3.2 Each (250/125)-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks,

2. The pilot cell specific gravity, corrected to (77)°F and full electrolyte level, is greater than or equal to ____, and

3. The pilot cell voltage is greater than or equal to ____ volts.

b. At least once per 92 days by verifying that:

1. The electrolyte level of each cell is between the minimum and maximum level indication marks,

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[SEE INSERTS "B" AND "C" ATTACHED]

SURVEILLANCE REQUIREMENTS (Continued)

2. The average specific gravity of each connected cell, corrected to (77)°F and full electrolyte level, is greater than or equal to (1.200),

3. The electrolyte temperatures in a representative sample of cells consisting of at least every sixth cell are within ± 5°F,

4. The minimum specific gravity, corrected to (77°)F and full electrolyte level, of each connected cell is within 0.010 of the average corrected value of all the connected cells,

5. The voltage of each connected cell is within ± 0.04 volts of the average voltage of all the connected cells,

6. The total battery terminal voltage is greater than or equal to ________ volts, and

7. The battery load (charger current) with the battery on float charge is less than ________ amps.

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 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 ______ ohms, and

4. The battery charger will supply at least ___ amperes at a minimum of ___ 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 emergency loads for (8) hours when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 90% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test.

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INSERT "A"

ELECTRICAL POWER SYSTEM

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D. C. bus train shall be OPERABLE and energized with an OPERABLE 125 volt D. C. bus, an OPERABLE full capacity charger, and an OPERABLE D.C. battery bank:

Channel "A", and
Channel "B", and
Channel "C" or
Channel "D"

APPLICABILITY: MODES 1, 2, 3, or 4

ACTION:

a. With one or more battery banks being inoperable as a result of being outside of the 7 day or 92 day surveillance requirements, operation may proceed for up to 7 days provided that all of the following are met:

   1. The electrolyte level of each cell is above the top of the plates and not overflowing.

   2. The average specific gravity of all the connected cells, corrected for temperature and electrolyte level, is not more than 20 points (0.020) below the manufacturer's recommended full charge specific gravity of 1.215 or the battery charging current is less than 2 amperes.

   3. The specific gravity, corrected for temperature and electrolyte level of each connected cell, is not more than 20 points (0.020) below the average corrected value of all the connected cells.

   4. The float voltage of each connected cell is greater than 2.07 volts.

   Otherwise restore the inoperable battery to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one 125 volt D.C. bus or battery charger inoperable, restore the inoperable bus or charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

INSERT "B"

ELECTRICAL POWER SYSTEM

D.C. DISTRIBUTION - OPERATING

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined to be energized and OPERABLE at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.3.2 Each 125 volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying all of the following:

1. The electrolyte level of each designated pilot cell is above the minimum level indication mark and not more than 1/4" above the maximum level indication mark

2. The specific gravity of each designated pilot cell corrected for temperature and electrolyte level, is greater than or equal to 1.200 or the battery charging current is less than 2 amperes

3. The float voltage of each designated pilot cell, corrected for temperature, is greater than or equal to 2.13 volts

4. The battery terminal voltage is greater than or equal to 129 volts on float charge, and

5. There is no evident physical damage to the racks, battery or connections that will limit battery operability

b. At least once per 92 days and within 7 days after a battery discharge, (battery terminal voltage less than 110 volts), battery overcharge (battery terminal voltage above 150 volts) or as a result of a battery pilot cell being outside of its 7-day surveillance requirements by verifying all the following:

1. The electrolyte level of each connected cell is above the minimum LEVEL indication mark and not more than 1/4" above the maximum level indication mark

2. The average specific gravity of all connected cells, corrected for temperature and electrolyte level, is not more than 10 points (0.010) below the manufacturer's recommended full charge specific gravity of 1.215.

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INSERT "C"

3. The specific gravity, corrected for temperature and electrolyte level, of each connected cell is not more than 20 points (0.020) below the manufacturer's recommended full charge specific gravity 1.215,

4. The average electrolyte temperature of a representative number of connected cells is above the minimum temperature 60° F,

5. The float voltage, corrected for the average temrperature, of each connected cell is equal to or greater than 2.13 volts, and,

6. There is no visible corrosion at either terminals or connectors or the connecting resistance of these affected item(s) is less than 150 micro OHMS.

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 intercell and terminal connection is less than or equal to 150 micro ohms, and,

4. The battery charger will supply at least 300 amperes at a minimum of 130 volts for at least 12 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the required time period 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 of the battery service test once per 60 month interval.

Performance tests of battery capacity should be given at least once per 18 months during COLD SHUTDOWN to any battery that shows signs of degradation or has reached 85 percent 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.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, one D.C. bus train consisting of the following shall be energized and OPERABLE with an operable 125 volt D.C. bus, an OPERABLE full capacity charger, and an OPERABLE D.C. battery bank.

~~1 - (250/125)-volt D.C. bus, and~~

~~1 - (250/125)-volt battery bank and full capacity charger associated with the above D.C. bus.~~

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required ~~(250~~/125)-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability with an overall voltage of greater than or equal to ~~(250~~/125) volts.

4.8.2.4.2 The above required ~~(250~~/125)-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

ELECTRICAL POWER SYSTEMS

3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

(LATER)

3.8.3.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the containment penetration conductor overcurrent protective devices shown in Table 3.8-1 inoperable:

a. Restore the protective device(s) to OPERABLE status or de-energize the circuits(s) by tripping the associated backup circuit breaker within 72 hours and verify the backup circuit breaker to be tripped at least once per 7 days thereafter; the provisions of Specification 3.0.4 are not applicable to overcurrent devices in circuits which have their backup circuit breakers tripped, or

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

SURVEILLANCE REQUIREMENTS

4.8.3.1 All containment penetration conductor overcurrent protective devices shown in Table 3.8-1 shall be demonstrated OPERABLE:

a. At least once per 18 months:

1. By verifying that the medium voltage (4-15 KV) circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers of each voltage level, and performing the following:

(a) A CHANNEL CALIBRATION of the associated protective relays, and

(b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and control circuits function as designed and as specified in Table 3.8.1.

SURVEILLANCE REQUIREMENTS (Continued)

(LATER)

(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. The functional test shall consist of injecting a current input at the specified setpoint to each selected circuit breaker and verifying that each circuit breaker functions as designed. Circuit breakers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

3. By selecting and functionally testing a representative sample of each type of fuse on a rotating basis. Each representative sample of fuses shall include at least 10% of all fuses of that type. The functional test shall consist of a non-destructive resistance measurement test which demonstrates that the fuse meets it manufacturer's design criteria. Fuses found inoperable during these functional tests shall be replaced with OPERABLE fuses prior to resuming operation. For each fuse found inoperable during these functional tests, an additional representative sample of at least 10% of all fuses of that type shall be functionally tested until no more failures are found or all fuses of that type have been functionally tested.

b. At least once per 60 months by subjecting each circuit breaker to an inspection and preventive maintenance in accordance with procedures prepared in conjunction with its manufacturer's recommendations.

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TABLE 3.8-1

(LATER)

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

Device Number and Location	Trip Setpoint (Amperes)	Response Time (sec/cycles)	System Powered
1. 6900 VAC (Primary breaker) (Back-up breaker)			Reactor Coolant pump 1 2 3 4
2. 480 VAC from MOAD Centers List all; primary breakers Back-up breakers " "			
3. 480 VAC from MCC List all; primary breakers Back-up breakers " "			
5. 440 VAC CEADM Power Primary breakers Back-up breakers " "			

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

(LATER)

3.8.3.2 The thermal overload protection and/or bypass devices, integral with the motor starter, of each valve listed in Table 3.8.2 shall be OPERABLE.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection and/or bypass devices inoperable, declare the affected valve(s) inoperable and apply the appropriate ACTION Statement(s) for the affected valve(s).

SURVEILLANCE REQUIREMENTS

4.8.3.2 The above required thermal overload protection and/or bypass devices shall be demonstrated OPERABLE:

a. At least once per 18 months, by the performance of a CHANNEL FUNCTIONAL TEST of the bypass circuitry for those thermal overload devices which are either:

   1. Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

   2. Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of all thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

(LATER)

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 ~~With the reactor vessel head unbolted or removed,~~ The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% delta k/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to ~~(1731)~~ 1720 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6~~*~~.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to {40} gpm of a solution containing ~~(1731)~~ greater than or equal to 1720 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 ~~(1731)~~ ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

~~*The reactor shall be maintained in MODE 6 whenever the reactor vessel head is unbolted or removed and fuel is in the reactor vessel.~~

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 ~~With the reactor vessel head unbolted or removed,~~ The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% delta k/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to ~~(1731)~~ 1720 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6~~*~~.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to {40} gpm of a solution containing ~~(1731)~~ 1720 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 ~~(1731)~~ ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

~~*The reactor shall be maintained in MODE 6 whenever the reactor vessel head is unbolted or removed and fuel is in the reactor vessel.~~

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

a. With one of the above required monitors inoperable, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

b. With both of the above required monitors inoperable, determine the boron concentration of the reactor coolant system at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least {72} hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than {72} hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least {72} hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment building penetrations shall be in the following status:

a. The equipment door closed and held in place by a minimum of four bolts,

b. A minimum of one door in each airlock is closed, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

1. Closed by an isolation valve, blind flange, or manual valve, or

2. Be capable of being closed by an OPERABLE automatic containment purge valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment purge valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment purge valves per the applicable portions of Specification (4.6.~~4~~ 3.2).

REFUELING OPERATIONS

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. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within one hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

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3/4.9.6 ~~MANIPULATOR CRANE~~ REFUELING MACHINE

LIMITING CONDITION FOR OPERATION

3.9.6 The ~~manipulator cranes~~ refueling machine shall be used for movement of CEAs or fuel assemblies and shall be OPERABLE with: one

a. A minimum capacity of ~~(2750) pounds, and~~ the combined normal weight of the fuel assembly, one part length CEA, grapple and refueling machine hoist, and

b. An automatic overload cut off limit. ~~of less than or equal to (2700) pounds.~~

APPLICABILITY: During movement of CEAs or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for ~~crane~~ the refueling machine OPERABILITY not satisfied, suspend ~~use of any inoperable manipulator crane from~~ all refueling machine operations involving the movement of CEAs and fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 The ~~manipulator crane~~ refueling machine used for movement of CEAs or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of ~~such~~ operations by performing ~~a~~ load tests of at least ~~(2750) pounds~~ the applicable minimum capacity and demonstrating an automatic load cut off when the ~~crane~~ refueling machine load exceeds ~~(2700) pounds.~~ applicable loads plus an administratively set margin.

refueling

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REFUELING OPERATIONS

3/4.9.~~8~~7 SHUTDOWN COOLANT AND COOLANT CIRCULATION

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.~~8~~7.1 At least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6

ACTION:

a. With less than one shutdown cooling loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The shutdown cooling loop may be removed from operation for up to 1 hour per 8 hour period. ~~during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.~~

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.~~8~~7 At least one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to ~~(3000)~~ (LATER) gpm at least once per ~~4~~ 12 hours.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.~~8~~7.2 Two independent shutdown cooling loops shall be OPERABLE.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

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

b. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.~~8~~7.2 The required shurdown cooling loops sall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each shutdown cooling loop.

REFUELING OPERATIONS

3/4.9.~~9~~8 CONTAINMENT PURGE ~~VALVE~~ ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.~~8~~8 The containment purge ~~valve~~ isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the containment purge ~~valve~~ isolation system inoperable, close each of the containment purge penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.~~8~~8 The containment purge ~~valve~~ isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge ~~valve~~ isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

REFUELING OPERATIONS

3/4.9.~~10~~ 9 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.~~10~~ 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 CEAs within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.~~10~~ 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 CEAs.

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3/4.9.~~11~~10 WATER LEVEL-STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.~~11~~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 storage pool.

ACTION:

With the requirement of the 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. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.~~11~~10 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

REFUELING OPERATIONS

3/4.9.~~12~~ 11 ~~STORAGE POOL AIR CLEANUP~~ FUEL HANDLING BUILDING EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.~~12~~ 11 Two independent fuel ~~storage pool air cleanup~~ handling building emergency ventilation systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel movement is in the storage pool is in progress.

ACTION:

a. With one fuel ~~storage pool air cleanup~~ handling building emergency ventilation system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the ~~OPERABLE~~ alternate fuel ~~storage pool air cleanup~~ handling building emergency ventilation system is ~~in operation and discharging through at least one train of HEPA filters and charcoal adsorbers~~ is demonstrated to be OPERABLE immediately and daily thereafter.

b. With no fuel ~~storage pool air cleanup~~ handling building emergency ventilation system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one ~~spent fuel storage pool~~ air cleanup system is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.~~12~~ 11 The above required ~~spent fuel storage pool air cleanup~~ fuel handling building emergency ventilation systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least ~~15 minutes (~~10 hours with the heaters on~~)~~, automatic.

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:

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SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 12925 cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is less than or equal to 1% when the system is tested by admitting cold DOP at the system intake. ~~(For systems with diverting valves.)~~

2. Verifying that the cleanup system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c and C.5.d of Regulatory Guide 1.52, Revision 2, March 1978, and the system flow rate is 12925 cfm ± 10%.

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

4. Verifying a system flow rate of 12925 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

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

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than (6) inches Water Gauge while operating the system at a flow rate of 12925 cfm ± 10%.

2. Verifying that on a high radiation test signal, the system automatically starts (unless already operating) and directs its exhaust flow through the HEPA filters and charcoal adsorber banks.

SURVEILLANCE REQUIREMENTS (Continued)

~~3. Verifying that the system maintains the spent fuel storage pool area at a negative pressure of greater than or equal to (1/4) inches Water Gauge relative to the outside atmosphere during system operation.~~

~~4. Verifying that the filter cooling bypass valves can be manually opened.~~

3 ~~5~~. {Verifying that the heaters dissipate 33 ± 1.7 kw when tested in accordance with ANSI N510-1975.}

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to {99.95}%* of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 925 cfm ± 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 12925 cfm ± 10%.

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

3/4.10 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 CEA worth and shutdown margin provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA{s}.

APPLICABILITY: MODE 2.

ACTION:

greater than or equal to 1720

a. With any full length CEA 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 {40} gpm of a solution containing ~~{1731}~~ ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

greater than or equal to 1720

b. With all full length CEAs inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at greater than or equal to {40} gpm of a solution containing ~~{1731}~~ ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA required either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each CEA not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 72 ~~24~~ hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications {3.1.1.3}, {3.1.3.1}, 3.1.3.2, {3.1.3.5}, {3.1.3.6}, ~~{3.1.3.8}~~, {3.2.2}, {3.2.3} and {the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1} may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed {85}% of RATED THERMAL POWER, and

b. The limits of Specification {3.2.1} are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification {3.2.1} being exceeded while any of the above requirements ~~of Specifications (3.1.1.3), (3.1.3.1), (3.1.3.5), (3.1.3.6), (3.1.3.8), (3.2.2), (3.2.3) and (the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1)~~ are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification (3.2.1), or

b. Be in HOT STANDBY wthin 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which any of the above ~~the~~ requirements ~~of Specifications (3.1.1.3), (3.1.3.1), (3.1.3.5), (3.1.3.6), (3.1.3.8), (3.2.2), (3.2.3) or (the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1) are suspended and shall be verified to be within the test power plateau.~~ are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification {3.2.1} by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications {4.2.1.3} and {3.3.3.2} during PHYSICS TESTS above 5% of RATED THERMAL POWER in which ~~the~~ any of the above requirements ~~of Specifications (3.1.1.3), (3.1.3.1), (3.1.3.5), (3.1.3.6), (3.1.3.8), (3.2.2), (3.2.3) or (the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1)~~ are suspended.

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification {3.4.1} and {noted requirements of Table 3.3-1} may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at ~~less than or equal~~ to {20}% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER ~~greater than~~ 5% of RATED THERMAL POWER, immediately trip the reactor.

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 startup and PHYSICS TESTS.

4.10.3.2 Each ~~wide range~~ logarithmic and linear power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.4 CENTER CEA MISALIGNMENT

LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.7 may be suspended during the performance of PHYSICS TESTS to determine the isothermal temperature coefficient, moderator temperature coefficient and power coefficient provided:

a. Only the center CEA (CEA #1) is misaligned, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.4.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.3.1 and 3.1.3.~~7~~ 6 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the requirements of Specification 3.2.1, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.~~7~~ 6 are suspended and shall be verified to be within the test power plateau.

4.10.4.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specification 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.~~7~~ 6 are suspended.

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NOTE

The ~~BASES~~ summary statements contained in this section provide the bases of the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specification.

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3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.1 requires each Reactor Coolant System safety injection tank to be OPERABLE and provides explicit ACTION requirements if one safety injection tank is inoperable. Under the terms of Specification 3.0.3, if more than one safety injection tank is inoperable, the Unit is required to be in at least HOT STANDBY within 1 hour and in at least HOT SHUTDOWN within the following 6 hours. As a further example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required Containment Spray Systems are inoperable, the unit is required to be in at least HOT STANDBY within 1 hour, in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN in the next 30 hours. It is assumed that the unit is brought to the required MODE within the required times by promptly initiating and carrying out the appropriate ACTION statement.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this specification 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.

3.0.5 This specification delineates what additional conditions must be satisfied to permit operation to continue, consistent with the ACTION statements for power sources, when a normal or emergency power source is not OPERABLE. It specifically prohibits operation when one division is inoperable because its normal or emergency power source is inoperable and a system, subsystem, train, component or device in another division is inoperable for another reason.

The provisions of this specification permit the ACTION statements associated with individual systems, subsystems, trains, components, or devices to be consistent with the ACTION statements of the associated electrical power source. It allows operation to be governed by the time limits of the ACTION statement associated with the Limiting Condition for Operation for the normal or emergency power source, not the individual ACTION statements for each system, subsystem, train, component or device that is determined to be inoperable solely because of the inoperability of its normal or emergency power source.

For example, Specification 3.8.1.1 requires in part that two emergency diesel generators be OPERABLE. The ACTION statement provides for a 72 hour out-of-service time when one emergency diesel generator is not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable emergency power source would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable Limiting Conditions for Operation. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable emergency diesel generator instead, provided the other specified conditions are satisfied. In this case, this would mean that the corresponding normal power source must be OPERABLE, and all redundant systems, subsystems, trains, components, and devices must be OPERABLE, or otherwise satisfy Specification 3.0.5 (i.e., be capable of performing their design function and have at least one normal or one emergency power source OPERABLE). If they are not satisfied, shutdown is required in accordance with this specification.

As a further example, Specification 3.8.1.1 requires in part that two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system be OPERABLE. The ACTION statement provides a 24-hour out-of-service time when both required offsite circuits are not OPERABLE. If the definition of OPERABLE were applied without consideration of Specification 3.0.5, all systems, subsystems, trains, components and devices supplied by the inoperable normal power sources, both of the offsite circuits, would also be inoperable. This would dictate invoking the applicable ACTION statements for each of the applicable LCOs. However, the provisions of Specification 3.0.5 permit the time limits for continued operation to be consistent with the ACTION statement for the inoperable normal power sources instead, provided theother specified conditions are satisfied. In this case, this would mean that for one division the emergency power source must be OPERABLE (as must be the components supplied by the emergency power source) and all redundant systems, subsystems, trains, components and devices in the other division must be OPERABLE, or likewise satisfy Specification 3.0.5 (i.e., be capable of performing their design functions and have an emergency power source OPERABLE). In other words, both emergency power sources must be OPERABLE and all redundant systems, subsystems, trains, components and devices in both divisions must also be OPERABLE. If these conditions are not satisfied, shutdown is required in accordance with this specification.

In MODES 5 or 6, Specification 3.0.5 is not applicable, and thus the individual ACTION statements for each applicable Limiting Condition for Operation in these MODES must be adhered to.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

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3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of ~~(5.0)~~ 5.15% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from any postulated accident are minimal and a ~~1~~ 2% delta k/k shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limitations on MTC are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

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 ~~(525)~~ 520°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 makeup pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of ~~1.0%~~ delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions ~~and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or (40,200)~~ 53,500 gallons of ~~(1731)~~ 1720 ppm borated water from the refueling water tank. (SEE INSERT "A", ATTACHED)

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 based upon providing a ~~1%~~ 2 delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either ~~(4,700)~~ 5465 gallons of ~~(1731)~~ 1720 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

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INSERT "A"

"...tank. However, for the purpose of consistency, the minimum required volume of (LATER) in Specification 3.1.2.8 is identical to the more restrictive value shown in Specification 3/4.5.4.

BORATION SYSTEMS (Continued)

The contained water volume limits includes 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.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between ~~(8.9)~~ 7.4 and ~~(11.0)~~ 8.0 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.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

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.

The ACTION statements applicable to a stuck or untrippable CEA to two or more inoperable CEAs and to a large misalignment (greater than or equal to 19 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 19 inches) of the CEAs, there is 1) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, 2) a small effect on the available SHUTDOWN MARGIN, and 3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

MOVABLE CONTROL ASSEMBLIES (Continued)

The CPCs provide protection to the core in the event of a large misalignment (greater than or equal to 19 inches) of a CEA by applying appropriate penalty factors to the calculation to account for the misaligned CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on 1) the available SHUTDOWN MARGIN, 2) the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints, and 3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA 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.

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The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to ~~(525)~~°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.

MOVABLE CONTROL ASSEMBLIES (Continued)

The establishment of LSSS and LCOs require that the expected long and short term behavior of the radial peaking factors be determined. The long term behavior relates to the variation of the steady state radial peaking factors with core burnup and is affected by the amount of CEA insertion assumed, the portion of a burnup cycle over which such insertion is assumed and the expected power level variation throughout the cycle. The short term behavior relates to transient perturbations to the steady-state radial peaks due to radial xenon redistribution. The magnitudes of such perturbations depend upon the expected use of the CEAs during anticipated power reductions and load maneuvering. Analyses are performed based on the expected mode of operation of the NSSS (base loaded, or load maneuvering). Analyses are performed based on the expected mode of operation of the NSSS (base load maneuvering, etc.) and from these analyses CEA insertions are determined and a consistent set of radial peaking factors defined. The Long Term Steady State and Short Term Insertion Limits are determined based upon the assumed mode of operation used in the analyses and provide a means of preserving the assumptions on CEA insertions used. The limits specified serve to limit the behavior of the radial peaking factors within the bounds determined from analysis. The actions specified serve to limit the extent of radial xenon redistribution effects to those accomodated in the analyses. The Long and Short Term Insertion Limits of Specification 3.1.3.6 are specified for the plant which has been designed for primarily base loaded operation but which has the ability to accomodate a limited amount of load maneuvering.

The Transient Insertion Limits of Specification 3.1.3.6 and the Shutdown CEA Insertion Limits of Specification 3.1.3.5 ensure that 1) the minimum SHUTDOWN MARGIN is maintained, and 2) the potential effects of a CEA ejection accident are limited to acceptable levels. Long term operation at the Transient Insertion Limits is not permitted since such operation could have effects on the core power distribution which could invalidate assumptions used to determine the behavior of the radial peaking factors.

~~(ALTERNATE)~~

~~The restriction prohibiting part length CEA insertion ensures that adverse power shapes and rapid local power changes which affect DNB considerations do not occur as a result of part length CEA insertion during operation.~~

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3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE (LATER)

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the Local Power Density channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core power operating limit corresponding to the allowable peak linear heat rate. Reactor operation at or below this calculated power level assures that the limits of Figure 3.1-1 are not exceeded.

The COLSS calculated core power and the COLSS calculated core power operating limits based on linear heat rate are continuously monitored and displayed to the operator. A COLSS alarm is annunciated in the event that the core power exceeds the core power operating limit. This provides adequate margin to the linear heat rate operating limit for normal steady state operation. Normal reactor power transients or equipment failures which do not require a reactor trip may result in this core power operating limit being exceeded. In the event this occurs, COLSS alarms will be annunciated. If the event which causes the COLSS limit to be exceeded results in conditions which approach the core safety limits, a reactor trip will be initiated by the Reactor Protective Instrumentation. The COLSS calculation of the linear heat rate limit includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the maximum linear heat rate calculated by COLSS is greater than or equal to that existing in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_r measurement uncertainty factor of 1.080, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for flux peaking augmentation and rod bow.

Parameters required to maintain the operating limit power level based on linear heat rate, margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-3 can be maintained by utilizing a predetermined local power density margin and a total core power limit in the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPCs.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.2 RADIAL PEAKING FACTORS (LATER)

Limiting the values of the planar radial peaking factors (F_r^c) used in the COLSS and CPCs to values equal to or greater than the measured planar radial peaking factors(F_r^m) provides assurance that the limits calculated by COLSS and the CPCs remain valid. Data from the incore detectors are used for determining the measured planar radial peaking factors. The periodic surveillance requirements for determining the measured planar radial peaking factors provides assurance that the planar radial peaking factors used in COLSS and the CPCs remain valid throughout the fuel cycle. Determining the measured planar radial peaking factors after each fuel loading prior to exceeding 70% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.3 AZIMUTHAL POWER TILT - T_q (LATER)

The limitations on the AZIMUTHAL POWER TILT are provided to ensure that design safety margins are maintained. An AZIMUTHAL POWER TILT greater than 0.10 is not expected and if it should occur, operation is restricted to only those conditions required to identify the cause of the tilt. The tilt is normally calculated by COLSS. The surveillance requirements specified when COLSS is out of service provide an acceptable means of detecting the presence of a steady state tilt. It is necessary to explicitly account for power asymmetries because the radial peaking factors used in the core power distribution calculations are based on an untilted power distribution.

AZIMUTHAL POWER TILT is measured by assuming that the ratio of the power at any core location in the presence of a tilt to the untilted power at the location is of the form:

$$P_{tilt}/P_{untilt} = 1 + T_q \, g \cos(\theta - \theta_o)$$

where:

T_q is the peak fractional tilt amplitude at the core periphery

g is the radial normalizing factor

θ is the azimuthal core location

θ_o is the azimuthal core location of maximum tilt

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AZIMUTHAL POWER TILT - T_q (Continued)

(LATER)

~~P_{tilt}/P_{untilt} is the ratio of the power at a core location in the presence of a tilt to the power at that location with no tilt.~~

3/4.2.4 DNBR MARGIN

(LATER)

The limitation on DNBR as a function of AXIAL SHAPE INDEX represents a conservative envelope of operating conditions consistent with the safety analysis assumptions and which have been analytically demonstrated adequate to maintain an acceptable minimum DNBR throughout all anticipated operational occurrences, of which the loss of flow transient is the most limiting. Operation of the core with a DNBR at or above this limit provides assurance that an acceptable minimum DNBR will be maintained in the event of a loss of flow transient.

Either of the two core power distribution monitoring systems, the Core Operating Limit Supervisory System (COLSS) and the DNBR channels in the Core Protection Calculators (CPCs), provide adequate monitoring of the core power distribution and are capable of verifying that the DNBR does not violate its limits. The COLSS performs this function by continuously monitoring the core power distribution and calculating a core operating limit corresponding to the allowable minimum DNBR. Reactor operation at or below this calculated power level assures that the limits of Figure 3.2-2 are not violated. The COLSS calculation of core power operating limit based on DNBR includes appropriate uncertainty and penalty factors necessary to provide a 95/95 confidence level that the core power at which a DNBR of less than 1.30 could occur, as calculated by COLSS, is less than or equal to that which would actually be required in the core. To ensure that the design margin to safety is maintained, the COLSS computer program includes an F_r measurement uncertainty factor of 1.080, an engineering uncertainty factor of 1.03, a THERMAL POWER measurement uncertainty factor of 1.02 and appropriate uncertainty and penalty factors for flux peaking augmentation and rod bow.

Parameters required to maintain the margin to DNB and total core power are also monitored by the CPCs. Therefore, in the event that the COLSS is not being used, operation within the limits of Figure 3.2-3 can be maintained by utilizing a predetermined DNBR as a function of AXIAL SHAPE INDEX and by monitoring the CPC trip channels. The above listed uncertainty and penalty factors are also included in the CPC.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.5 RCS FLOW RATE

(LATER)

~~This specification is provided to ensure that the actual RCS total flow rate is maintained at or above the minimum value used in the LOCA safety analyses.~~

3/4.2.6 CORE AVERAGE COOLANT TEMPERATURE

(LATER)

~~This specification is provided to ensure that the assumptions used for the initial conditions of the LOCA safety analyses remain valid.~~

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the reactor protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the 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.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix "A" of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs," February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975.

~~3/4.3.3.7 CHLORINE DETECTION SYSTEMS~~

~~The OPERABILITY of the chlorine detection system ensures that sufficient capability is available to promptly detect and initiate protective action in the event of an accidental chlorine release. This capability is required to protect control room personnel and is consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release," February 1975.~~

3/4.3.3.~~8~~ 7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODES 4 and 5, a single reactor coolant loop or shutdown cooling ~~loop~~ train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two ~~loops~~ trains be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling ~~loops~~ trains to be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling 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 reductions will, therefore, be within the capability of operator recognition and control.

(LATER)

~~The restrictions on starting a Reactor Coolant Pump during MODES 4 and 5 with one or more RCS cold legs less than or equal to (275)°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary 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 (46)°F above each of the RCS cold leg temperatures.~~

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 4.6×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System (see Specification 3/4.4.9.3) provides a diverse means of protection against RCS overpressurization at low temperatures.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves ~~and power operated relief valve~~ against water relief. The requirement that a minimum number of pressurizer heaters be OPERABLE assures that the plant will be able to establish natural circulation.

~~3/4.4.5 RELIEF VALVES~~

~~The power operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs in conjunction with a reactor trip on a Pressurizer Pressure High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable.~~

3/4.4.~~6~~ 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.

STEAM GENERATORS (Continued)

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 = ~~(0.5)~~ 1.0 GPM 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 ~~(0.5)~~ 1.0 GPM per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of ~~(40)~~ (LATER)% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.~~X~~6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.~~X~~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

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Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.~~7~~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 allowances 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 ( ) GPM with the modulating valve in the supply line fully open at a nominal RCS pressure of (2230) psig. This limitation ensures that in the event of a LOCA, the safety injection flow will not be less than assumed in the accident analyses.~~

The surveillance requirements for RCS Pressure Isolation Valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS Pressure Isolation Valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents. The ~~(0.5)~~ 1.0 GPM leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.~~8~~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

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.~~9~~ 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 and a concurrent loss of offsite electrical power. 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 ~~( )~~ SAN ONOFRE 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~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding ~~1.0~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2 hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than ~~1.0~~ 6.5 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800 hour limit.

SPECIFIC ACTIVITY (Continued)

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.~~10~~ 9 PRESSURE/TEMPERATURE LIMITS (LATER)

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section (   ) of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

PRESSURE/TEMPERATURE LIMITS (Continued) (LATER)

The heatup and cooldown limit curves (Figure 3.4-2) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to (75)°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table (B 3/4.4-1). Reactor operation and resultant fast neutron (E greater than 1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figure (B 3/4.4-1). The heatup and cooldown limit curves Figures (3.4-2) include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure (3.4-2) for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be (50)°F. The Lowest Service Temperature limit line shown on Figure (3.4-2) is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of ____ psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch @ 30 ft - lb	Temperature of Charpy V-Notch @ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb

(LATER)

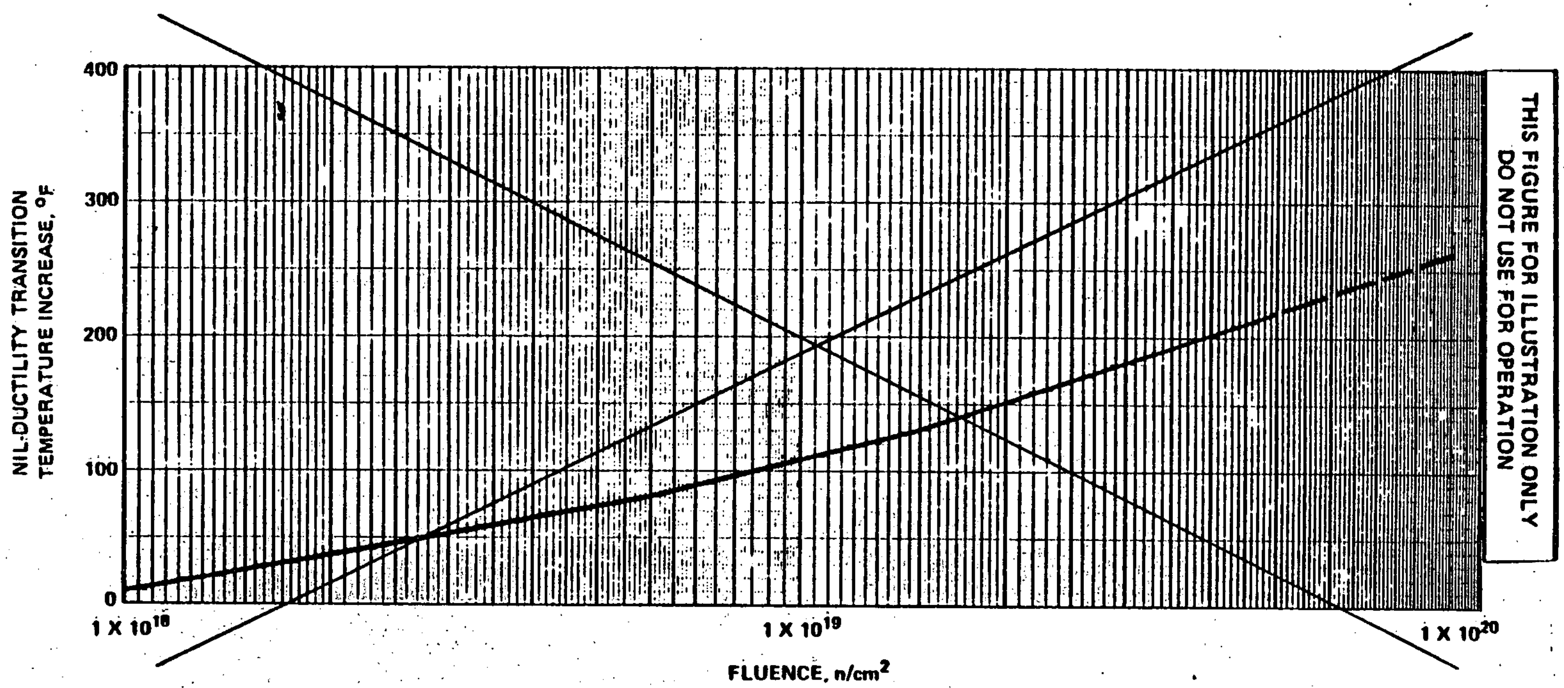

Figure B3/4.4-1
Nil-Ductility Transition Temperature Increase
as a Function of Fast ($E > 1$mev) Neutron Fluence (550°F Irradiation)

(LATER)

PRESSURE/TEMPERATURE LIMITS (Continued) the Shutdown Cooling System relief valve

The Shutdown Cooling System Relief Valve

The OPERABILITY of ~~two PORVS~~ or a RCS vent opening of greater than (1.3) square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to (275)°F. ~~Either PORV~~ has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to (46)°F above the RCS cold leg temperatures or (2) the start of a HPSI pump and its injection into a water solid RCS.

3/4.4.~~11~~ 10 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition and Addenda through Summer 1975.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks 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 safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The safety injection tank 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 safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and 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 safety injection tanks 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.

EMERGENCY CORE COOLING SYSTEMS

BASES

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below ~~300~~ 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.

[SEE INSERT "A", ATTACHED]

~~The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the prossibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.~~

~~The limitation for a maximum of one high pressure safety injection pump to be OPERABLE, and the Surveillance Requirement to verify all high pressure safety injection pumps except the required OPERABLE pump to be inoperable below (275)°F, provides assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.~~

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident 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. ~~The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post LOCA temperatures.~~

3/4.5.4 REFUELING WATER STORAGE TANKS ~~(RWT)~~ RWST's

The OPERABILITY of the ~~RWT~~ RWST's as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on ~~RWT~~ RWST's minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the ~~RWT~~ RWST's and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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INSERT "A"

The NaOH added to the Containment Spray, via the Spray Chemical Addition pumps, minimizes the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The NaOH additive results in the final pH being raised to greater than or equal to 7.0.

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REFUELING WATER STORAGE TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between ~~(8.9)~~ 8.0 and ~~(11.0)~~ 10.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

BASES

FOR

SECTION 3/4.6~~F~~

CONTAINMENT SYSTEMS SPECIFICATIONS

~~FOR~~

~~COMBUSTION ENGINEERING~~

~~ATMOSPHERIC TYPE CONTAINMENT~~

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3/4.6 CONTAINMENT SYSTEMS

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

~~3/4.6.1.4 CONTAINMENT ISOLATION VALVE AND CHANNEL WELD PRESSURIZATION SYSTEMS (OPTIONAL)~~

~~The OPERABILITY of the isolation valve and containment channel weld pressurization systems is required to meet the restrictions on overall containment leak rate assumed in the accident analyses. The Surveillance Requirements for determining OPERABILITY are consistent with Appendix J of 10 CFR 50.~~

3/4.6.1.~~8~~ 4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of ~~(3.0)~~ 5.0 psig and 2) the containment peak pressure does not exceed the design pressure of ~~(54)~~ 60.0 psig during ~~(LOCA or~~ steam line break conditions~~)~~.

The maximum peak pressure expected to be obtained from ~~a (LOCA or~~ steam line break~~)~~ event is ~~(45)~~ 55.7 psig. The limit of ~~(3)~~ 1.5 psig for initial positive containment pressure will limit the total pressure to ~~(48)~~ 57.2 psig which is less than the design pressure and is consistent with the accident analyses.

3/4.6.1.~~6~~ 5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a ~~(LOCA or~~ steam line break accident~~)~~.

3/4.6.1.~~7~~ 6 CONTAINMENT STRUCTURAL INTEGRITY

~~(Prestressed concrete containment with ungrouted tendons.)~~

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of ~~(48)~~ 55.7 psig in the event of a ~~(LOCA or~~ steam line break accident~~)~~. The measurement of containment tendon lift off force, the tensile tests of the tendon wires or strands, the visual examination of tendons, anchorages and exposed interior and exterior surfaces of the containment and the Type A leakage tests are sufficient to demonstrate this capability. (The tendon wire or strand samples will also be subjected to stress cycling tests and to accelerated corrosion tests to simulate the tendon's operating conditions and environment.)

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures", January 1976.

~~(Reinforced concrete containment.)~~

~~This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment~~

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CONTAINMENT SYSTEMS

BASES

CONTAINMENT STRUCTURAL INTEGRITY (Continued)

will withstand the maximum pressure of (48) psig in the event of a (LOCA or steam line break accident). A visual inspection in conjunction with the Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.8 CONTAINMENT VENTILATION SYSTEM

The 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.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the containment spray system ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

~~(Credit taken for iodine removal)~~

The containment spray system and the containment cooling system are redundant to each other in providing post accident cooling of the containment atmosphere. However, the containment spray system also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

(No credit taken for iodine removal)

The containment spray system and the containment cooling system are redundant to each other in providing post accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the containment spray system, the allowable out of service time requirements for the containment spray system and containment cooling system have been interrelated and adjusted to reflect this additional redundancy in cooling capability.

3/4.6.2.2 ~~SPRAY ADDITIVE~~ IODINE REMOVAL SYSTEM ~~(OPTIONAL)~~

The OPERABILITY of the ~~spray additive~~ iodine removal system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH

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IODINE REMOVAL
~~SPRAY ADDITIVE~~ SYSTEM (Continued)

volume and concentration ensure a pH value of between 8.0 ~~(8.9)~~ and 10.0 ~~(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 stress corrosion the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the accident analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM ~~(OPTIONAL)~~

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

~~(Credit taken for iodine removal by spray systems)~~

The containment cooling system and the containment spray system are redundant to each other in providing post accident cooling of the containment atmosphere. As a result of this redundancy in cooling capability, the allowable out of service time requirements for the containment cooling system have been appropriately adjusted. However, the allowable out of service time requirements for the containment spray system have been maintained consistent with that assigned other inoperable ESF equipment since the containment spray system also provides a mechanism for removing iodine from the containment atmosphere.

~~(No credit taken for iodine removal by spray systems)~~

~~The containment cooling system and the containment spray system are redundant to each other is providing post accident cooling of the containment atmosphere. Since no credit has been taken for iodine removal by the containment spray system, the allowable out of service time requirements for the containment cooling system and containment spray system have been interrelated and adjusted to reflect this additional redundancy in cooling capacity.~~

~~3/4.6.3 IODINE CLEANUP SYSTEM (OPTIONAL)~~

~~The OPERABILITY of the containment iodine filter trains ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site~~

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~~IODINE CLEANUP SYSTEM (Continued)~~

~~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.~~

~~(Cumulative operation of the system with the heaters on for at least 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.)~~

3/4.6.~~4~~3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.~~5~~4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit ~~(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~~ containment dome air circulators are provided to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

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

~~The OPERABILITY of the penetration room exhaust system ensures that radioactive materials leaking from the containment atmosphere through containment penetrations following a LOCA are filtered and adsorbed prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the LOCA analyses.~~

~~(Cumulative operation of the system with the heaters on for at least 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.)~~

3/4.6.7 VACUUM RELIEF VALVES (OPTIONAL)

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure does not become more negative than ____ psi. This condition is necessary to prevent exceeding the containment design limit for internal pressure differential of (1.0) psi.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% ~~(1100 psig)~~ of its design pressure of ~~(1000)~~ 1100 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 Vessel Code, ~~1971~~ 1974 Edition. The total relieving capacity for all valves on all of the steam lines is ~~( )~~ 15,130,000 lbs/hr which is ~~( )~~ 102.3 percent of the total secondary steam flow of ~~( )~~ 15,473,628 lbs/hr at 100% RATED THERMAL POWER. A minimum of ~~(2)~~ 1 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

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 Level-High channels. The reactor trip setpoint reductions are derived on the following bases: (SEE INSERT "A," ATTACHED)

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (125)$$

For single loop operation (two reactor coolant pumps operating in the same loop)

$$SP = \frac{(X) - (Y)(U)}{X} \times (**)$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

INSERT "A"

For two loop, four pump operation

$$f \leq \frac{(2)(N)(\dot{\omega}_{sv})(h_g - h_{fw})}{(Q)}$$

where:

f = maximum allowable fractional power level as a fraction of RATED THERMAL POWER ($f \leq 1.00$)

N = minimum number of operable main steam safety valves on any one generator

$\dot{\omega}_{sv}$ = steam flow capacity of each main steam safety valve at 1140 lb/in.^{2}g (lbm/h)

Q = secondary heat transfer rate of both generators at RATED THERMAL POWER (Btu/h)

h_g = enthalpy of the saturated steam at the operating pressure of the steam generators (Btu/lbm)

h_{fw} = feedwater enthalpy at RATED THERMAL POWER, which is assumed to remain constant in order to yield a more conservative power level (Btu/lbm)

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~~SAFETY VALVES (Continued)~~

U	=	maximum number of inoperable safety valves per operating steam line
(125)	=	Power Level-High Trip Setpoint for two loop operation
(**)	=	Power Level-High Trip Setpoint for single loop operation with two reactor coolant pumps operating in the same loop
X	=	Total relieving capacity of all safety valves per steam line in lbs/hour (lbs/hr)
Y	=	Maximum relieving capacity of any one safety valve in lbs/hour (lbs/hr)

3/4.7.1.2 ~~EMERGENCY~~ FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than {350}°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of ~~(350)~~ 700 gpm at a pressure of ~~(1100)~~ 1170 psig to the entrance of the steam generators. The steam driven auxiliary feedwater pump is capable of delivering a total feedwater flow of {700} gpm at a pressure of ~~(1100)~~ 1170 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than {350}°F when the shutdown cooling 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 ~~( )~~ 24 hours with steam discharge to atmosphere with concurrent with total loss of off-site power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

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3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVE

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations ~~to (90)~~ of 70°F and ~~(275)~~ 200 psig are based on a steam generator RT_{NDT} of 30°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~~ SALT WATER COOLING SYSTEM

The OPERABILITY of the service water system ensures that sufficient cooling capacity is available for continued operation of 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.

3/4.7.5 ULTIMATE HEAT SINK (OPTIONAL)

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants", March 1974.

3/4.7.6 FLOOD PROTECTION (OPTIONAL)

The limitation on flood protection ensures that facility protective actions will be taken (and operation will be terminated) in the event of flood conditions. The limit of elevation (__) Mean Sea Level is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.

3/4.7.~~7~~5 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the control room emergency air cleanup 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 expsoure 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.

Cumulative operation of the system with the heaters on for at least 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.7.8 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM

The OPERABILITY of the ECCS pump room exhaust air cleanup system ensures that radioactive materials leaking from the ECCS equipment within the pump room following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses. (Cumulative operation of the system with the heaters on for at least 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.)

3/4.7.~~9~~ 6 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Selection of a representative sample according to the expression $35\ (1 + \frac{c}{2})$ provides a confidence level of approximately 95% that 90% to 100% of the snubbers in the plant will be OPERABLE within acceptance limits. Observed failures of these sample snubbers shall require functional testing of additional units.

Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

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The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.

3/4.7.~~10~~ 7 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 shield mechanism.

3/4.7.~~11~~ 8 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression system consists of the water system, spray and/or sprinklers, ~~CO_2, Halon;~~ and fire hose stations. ~~and yard fire hydrants.~~ The collective capability of the fire suppression systems is adequate to minimize potential damage to safety related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The surveillance requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met. ~~An allowance is made for ensuring a sufficient volume of Halon in the Halon storage tanks by verifying either the weight or the level of the tanks. Level measurements are made by either a U.L. or F.M. approved method.~~

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.~~12~~ 9 FIRE BARRIER PENETRATIONS

The functional integrity of the fire barrier penetrations ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The fire barrier penetrations are a passive element in the facility fire protection program and are subject to periodic inspections.

Fire barrier penetrations, including cable penetration barriers, fire doors and dampers are considered functional when the visually observed condition is the same as the as-designed condition. For those fire barrier penetrations that are not in the as-designed condition, an evaluation shall be performed to show that the modification has not degraded the fire rating of the fire barrier penetration.

During periods of time when a barrier is not functional, either, 1) a continuous fire watch is required to be maintained in the vicinity of the affected barrier, or 2) the fire detectors on at least one side of the affected barrier must be verified OPERABLE and an hourly fire watch patrol established, until the barrier is restored to functional status.

~~3/4.7.13 AREA TEMPERATURE MONITORING~~

~~The area temperature limitations ensure that safety related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ( )°F.~~

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3/4.8 ELECTRICAL POWER SYSTEMS

BASES

3/4.8.1 and 3/4.8.2 A.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 Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the 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 OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are in accordance with the recommendations of ~~Regulatory Guide 1.129 "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978.~~

[SEE INSERT "A", ATTACHED]

3/4.8.3 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.

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INSERT "A"

". . . recommendations of . . . IEEE Standard 450-1975, Proposed Revision, Draft 4, "Recommended Practice for Maintenance, Testing, and Replacement of Large Stationary Type Power Plant and Substation Lead Storage Batteries."

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ELECTRICAL EQUIPMENT PROTECTIVE DEVICES (Continued)

The surveillance requirements applicable to lower voltage circuit breakers and fuses provides assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturers brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

The OPERABILITY of the motor operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 ~~MANIPULATOR CRANE~~ REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: 1) ~~manipulator cranes~~ refueling machines will be used for movement of CEAs and fuel assemblies, 2) each ~~crane~~ machine has sufficient load capacity to lift a CEA or fuel assembly, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

~~3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING~~

~~The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.~~

3/4.9.~~8~~7 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling 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 shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.~~9~~8 CONTAINMENT PURGE VALVE ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.~~10~~ and 3/4.9.~~11~~ WATER LEVEL-REACTOR VESSEL AND STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.~~12~~ 11 FUEL HANDLING BUILDING EMERGENCY VENTILATION UNIT ~~STORAGE POOL AIR CLEANUP SYSTEM~~

The limitations on the storage pool air cleanup system ensure that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

Cumulative operation of the system with the heaters on for at least 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

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3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs 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 CEAs 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 CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 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.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

SECTION 5.0

DESIGN FEATURES

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5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure ~~{~~5.1-1~~}~~.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure ~~{~~5.1-2~~}~~.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 150 feet.

b. Nominal inside height = 172 feet.

c. Minimum thickness of concrete walls = 4 1/3 feet.

d. Minimum thickness of concrete roof = 3 3/4 feet.

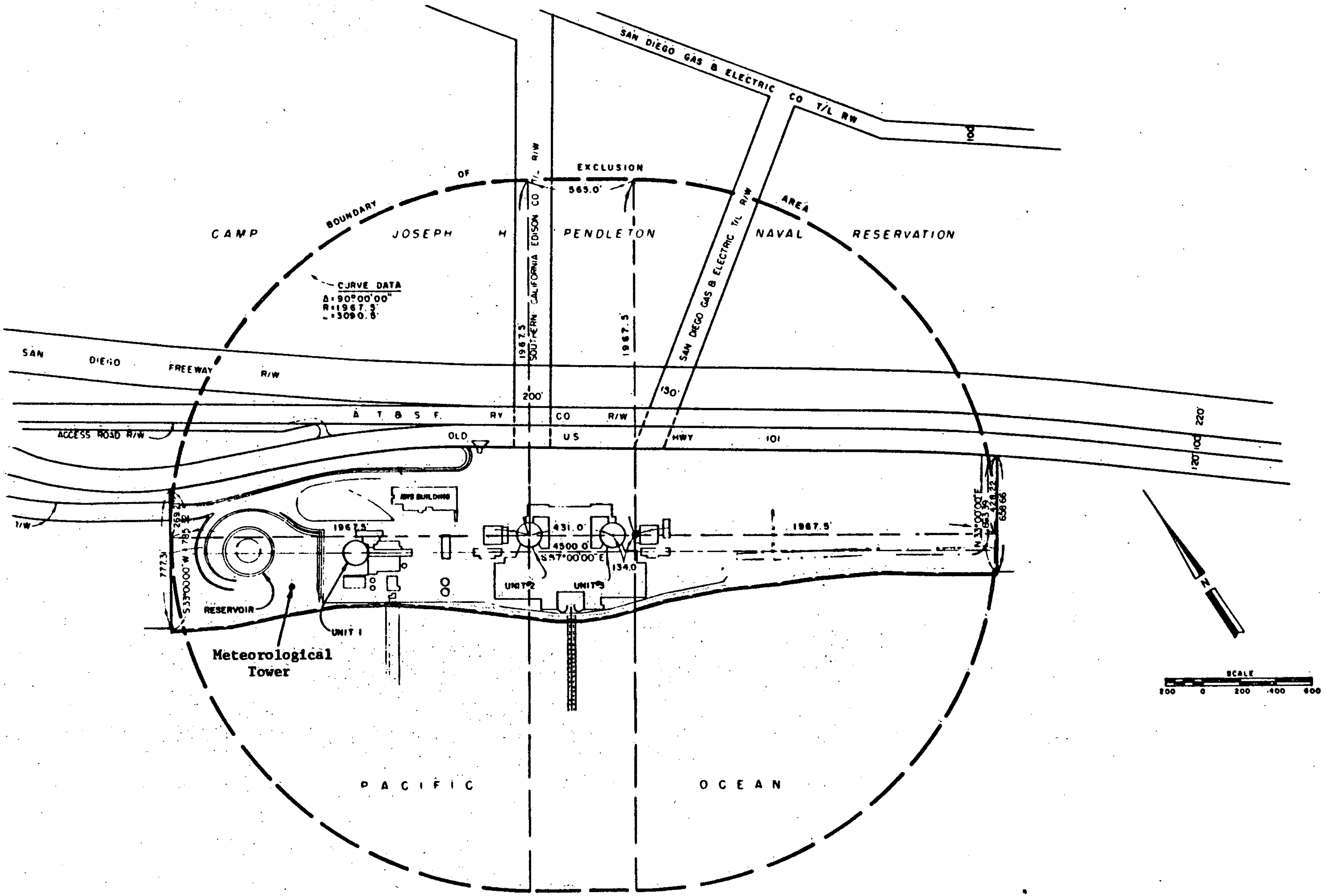
e. Minimum thickness of concrete floor pad = 9 feet.

f. Nominal thickness of steel liner = 1/4 inches.

g. Net free volume = 2,335,000 cubic feet.

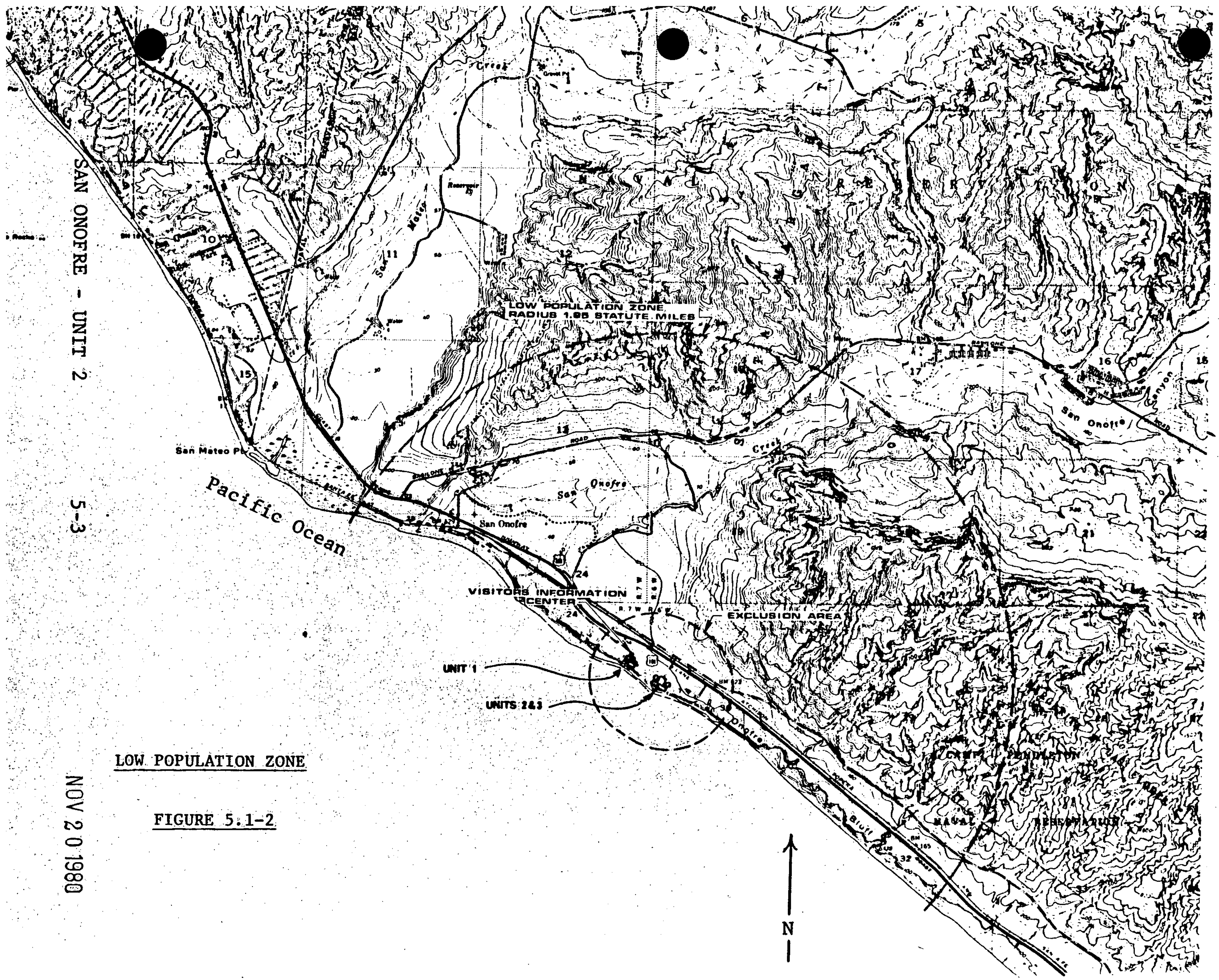
DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 60 psig and a temperature of 300°F.



EXCLUSION AREA

FIGURE 5.1-1



LOW POPULATION ZONE

FIGURE 5.1-2

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5.3 REACTOR CORE

FUEL ASSEMBLIES

a maximum of

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing 236 fuel rods clad with ~~{~~Zircaloy-4~~}~~. Each fuel rod shall have a nominal active fuel length of 150 inches and contain a maximum total weight of 1807 grams uranium. The initial core loading shall have a maximum enrichment of 2.88 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of ~~—~~ weight percent U-235.
(LATER)

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 full length and 8 part length control element assemblies.

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 of the applicable Surveillance Requirements,

b. For a pressure of 2500 psia, and

c. For a temperature of 650 °F, except for the pressurizer which is 700 °F, ~~°F.~~

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VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,800 ~~±~~ ___ +600/-0 cubic feet at a nominal T_{avg} of ~~(525)~~ 582.1°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 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance of ~~(2.6%)~~ 0.014 delta k/k for uncertainties as described in Section ~~(4.3)~~ 9.1 of the FSAR.

b. A nominal ~~(21)~~ 12.75 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.2 The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed ~~(~~0.98~~)~~ when aqueous foam moderation is assumed.

DRAINAGE

5.6.3 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 61 ft-6 in.

CAPACITY

5.6.4 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 800 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - T_{avg} from ≤ ~~200~~ 70°F to ≥ 545°F; cooldown cycle - T_{avg} from ≥ 545°F to ≤ ~~200~~ 70°F.
	500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr.	~~Heat~~ Heatup cycle - Pressurizer temperature from ≤ ~~200~~ 70°F to ≥ 653°F; cooldown ≥ 653°F to ≤ ~~200~~ 70°F
	10 hydrostatic testing cycles.	RCS pressurized to ~~3110 psig~~ 3125 psia with RCS temperature ≥ ~~60~~ 100°F. ~~above the most limiting components' NDTT value.~~
	200 leak testing cycles.	RCS pressured to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than minimum RCS temperature for critically.
	~~(400)~~ reactor trip cycles.	Trip from 100% of RATED THERMAL POWER.
	~~(40) turbine trip cycles with delayed reactor trip.~~	~~Turbine trip (total load rejection) from 100% of RATED THERMAL POWER followed by resulting reactor trip.~~
	200 seismic stress cycles.	Subjection to a seismic event equal to one half the design basis earthquake (DBE).

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	40 complete loss of reactor coolant flow cycles.	Simultaneous loss of all Reactor Coolant Pumps at 100% of RATED THERMAL POWER.
	5 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODES 1, 2 or 3.
	100 pressurizer spray cycles per year with pressurizer/spray water $\Delta T > 200°F$ or as otherwise calculated by the following method:	Spray operation consisting of opening and closing either the main or auxiliary spray valves(s) spray water/pressurizer $\Delta T > 200°F$.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Method for Calculating Pressurizer Spray Nozzle Cumulative Usage Factor

ΔT	N_A	N	N/N_A
201 - 300	13,000		
301 - 400	5,000		
401 - 500	3,000		
501 - 600	1,500		
		$\Sigma N/N_A$	

Where:

ΔT = Temperature difference between pressurizer water and spray in °F.

N_A = Allowable number of spray cycles.

N = Number of cycles in ΔT range indicated.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Calculational Method:

1. At 12 month intervals the cumulative spray cycles shall be totaled. If the total is equal to or less than 1000, no further action is required.

2. If the cumulative total exceeds 1000, the spray nozzle usage factor shall be calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the calculated usage factor is equal to or less than 0.75, no further action is required.

3. If the calculated usage factor exceeds 0.75, subsequent pressurizer spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 200°F when spray is operated. An engineering evaluation of nozzle fatigue shall be performed and shall determine that that the nozzle remains acceptable for additional service prior to removing this restriction.

SECTION 6.0

ADMINISTRATIVE CONTROLS

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SAN ONOFRE - UNIT 2

6.1 RESPONSIBILITY

6.1.1 The Manager ~~{Plant Superintendent}~~ the Vice-President of Nuclear 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 or during his absence from the Control Room designated individual shall be responsible for the Control Room command function and shall be the only individual that may direct the licensed activities of licensed operators. A management directive to this effect, signed by ~~the highest level of corporate management~~ shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown on Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition at least one licensed Senior Reactor Operator shall be in the Control Room while the Unit is in MODES 1,2,3 or 4

~~c. At least two licensed Reactor Operators shall be present in the control room during reactor startup, scheduled reactor shutdown and during recovery from reactor trips.~~

c ~~d~~. A ~~health physics~~ chemical-radiation protection technician# shall be on site when fuel is in the reactor.

d ~~e~~. 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 ~~f~~. A site Fire Brigade of at least 5 members shall be maintained onsite at all times#. The Fire Brigade shall not include (3) members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

#The ~~health physics~~ chemical-radiation protection technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accommodate unexpected absence provided immediate action is taken to fill the required positions.

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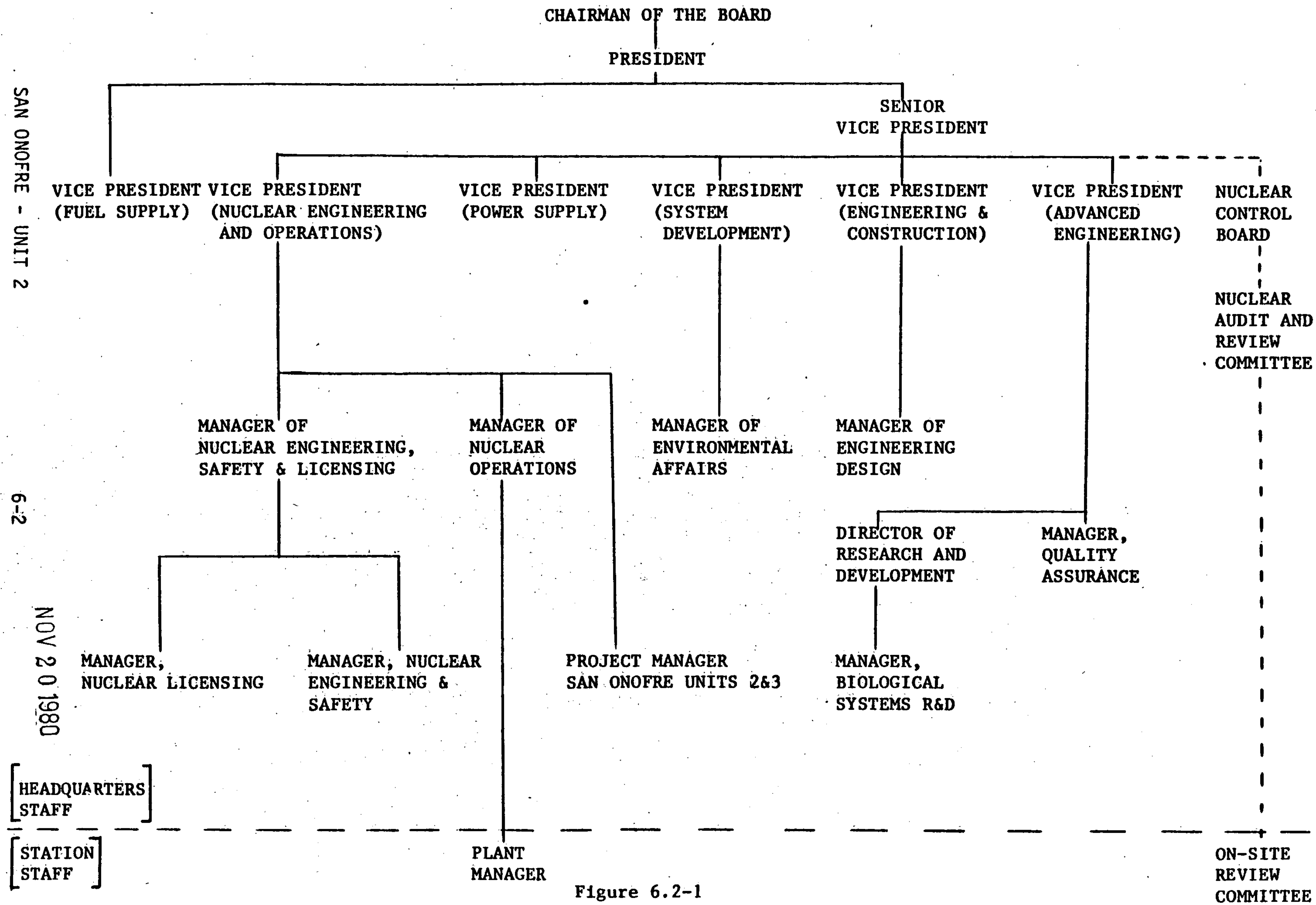
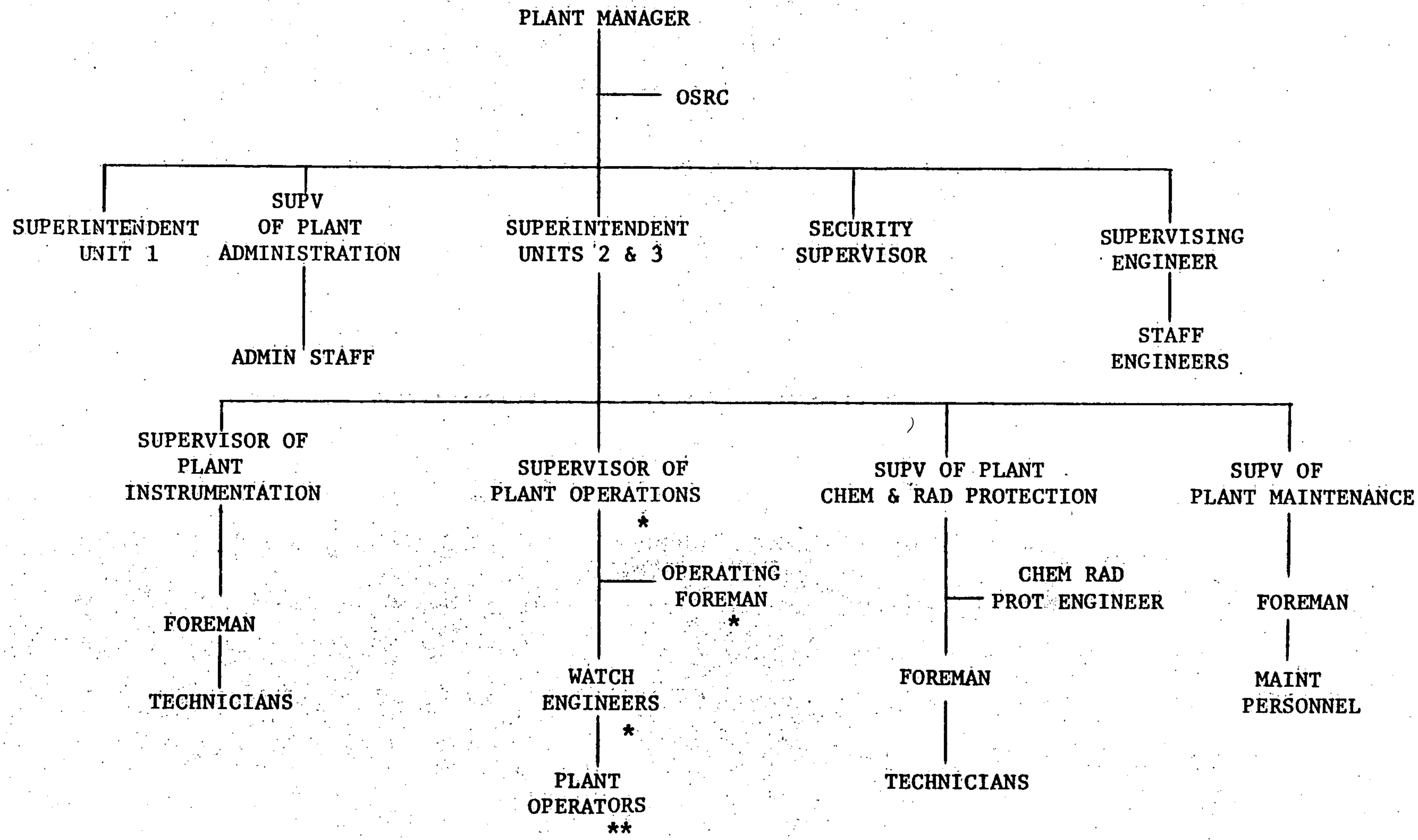


Figure 6.2-1

OFFSITE ORGANIZATION



* Senior Reactor Operator License Required
** Control and Assistant Control Operators are holders of Reactor Operator Licenses

Figure 6.2-2
UNIT ORGANIZATION

During any absence of the ~~Shift Supervisor~~ from the Control Room while the unit is in MODE 1,2,3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absense of the ~~Shift Supervisor~~ from the Control Room while the unit is in MODE 5 or 6, an individual with a valid RO license (other than the Shift Technical Advisor) shall be designated to assume the Control Room command function.

WATCH ENGINEER WATCH ENGINEER

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION

~~SINGLE UNIT FACILITY~~

POSITION	MODES 1,2,3 & 4	MODES 5 & 6
~~SS~~ WATCH ENGINEER	1	1
SRO*	1	** None Required
RO	2	1
Non-Licensed Auxiliary Operator	2	1
Shift Technical Advisor	1	None Required

Except for the ~~Shift Supervisor~~, 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.

Licensed operators shall not work more than:*

1. 12 hours straight
2. 24 hrs. in any 48 hr. period
3. 72 hours in any 7-day period
4. 14 consecutive days without having 2 consecutive days off.

*Deviation from these requirements may be authorized by the Plant Manager ~~Superintendent~~ in accordance with established procedures and with documentation of the cause. Overtime limits do not include shift turnover time.

** [SEE INSERT "A", ATTACHED]

INSERT "A"

**

In Mode 6, shift crew assignments during periods of core alterations shall include a licensed senior reactor operator (SRO) to directly supervise the core alterations. This licensed senior reactor operator may have fuel handling duties but shall not have other concurrent operational duties.

6.2.3 NUCLEAR EXPERIENCE REVIEW ~~PANEL (...)~~
~~6.2.3.1 The (NERP) shall be a multi-discipline review group and shall review nuclear industry operational experience~~
~~6.2.3.2 The (NERP) shall inform the Shift Technical Advisor of all such experience that is relevant to operation of the unit~~

ADMINISTRATIVE CONTROLS [SEE INSERT "A", ATTACHED]

6.3 UNIT STAFF QUALIFICATIONS

~~Minimum qualifications for members of the unit staff may be specified by use of an overall qualification statement referencing ANSI N18.1-1971 or alternately by specifying individual position qualifications. Generally, the first method is preferable; however, the second method is adaptable to those unit staffs requiring special qualification statements because of a unique organizational structure.~~

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 (1) the (Radiation Protection Manager) who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design, and response and analysis of the plant for transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the ~~(position title)~~ Plant Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55. ~~and~~ The program shall include familiarization with the relevant operational experience identified in the (NERP) and ~~shall include~~ degraded core training.

6.5 REVIEW AND AUDIT

~~The method by which independent review and audit of facility operations is accomplished may take one of several forms. The licensee may either assign this function to an organzational unit separate and independent from the group having responsibility for unit operation or may utilize a standing committee composed of individuals from within and outside the licensee's organization.~~

~~Irrespective of the method used, the licensee shall specify the details of each functional element provided for the independent review and audit process as illustrated in the following example specifications.~~

6.5.1 ~~UNIT REVIEW GROUP (URG)~~ ONSITE REVIEW COMMITTEE (OSRC)

FUNCTION

6.5.1.1 The ~~(Unit Review Group)~~ shall function to advise the (Plant ~~Superintendent)~~ Manager on all matters related to nuclear safety.

6.2.4 ~~SHIFT TECHNICAL ADVISOR~~ WATCH ENGINEER
6.2.4.1 The Shift Technical ~~Advisor~~ shall serve in an advisory capacity ~~to the Shift Supervisor~~ on matters pertaining to the engineering aspects of assuring safe operation of the unit.
~~6.2.4.2 The Shift Technical Advisor shall disseminate relevant operational experience identified by the (NERP)~~

INSERT "A"

6.2.3.1 The Manager, Nuclear Engineering and Safety shall be responsible for the review of operational experiences of the nuclear industry.

6.2.3.2 The results of the reviews of 6.2.3.1 above which are relevant to the operation of the unit shall be provided to the Manager of Nuclear Operations for dissemination within the operating organization.

COMPOSITION

6.5.1.2 The ~~(Unit Review Group)~~ Onsite Review Committee shall be composed of the:

(SEE INSERT "A", ATTACHED)

~~Chairman: (Plant Superintendent)~~
~~Member: (Operations Supervisor)~~
~~Member: (Technical Supervisor)~~
~~Member: (Maintenance Supervisor)~~
~~Member: (Plant Instrument and Control Engineer)~~
~~Member: (Plant Nuclear Engineer)~~
~~Member: (Health Physicist)~~

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the ~~(URG)~~ OSRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ~~(URG)~~ OSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The ~~(URG)~~ OSRC shall meet at least once per calendar month and as convened by the ~~(URG)~~ OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the ~~(URG)~~ OSRC necessary for the performance of the ~~(URG)~~ OSRC responsibility and authority provisions of these technical specifications shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The ~~(Unit Review Group)~~ Onsite Review Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant ~~Superintendent)~~ Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to ~~Appendix "A"~~ the Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

INSERT "A"

Chairman:	Plant Manager
Member:	Plant Superintendent
Member:	Supervisor of Plant Operations
Member:	Technical Supervisor
Member:	Supervisor of Plant Maintenance
Member:	Plant Instrument and Control Engineer
Member:	Plant Nuclear Engineer
Member:	Chemical and Radiation Protection Engineer
Member:	SDG&E Representative

Manager of Nuclear Operations

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the ~~(Superintendent of Power Plants)~~ and to the ~~(Company Nuclear Review and Audit Group).~~ (NARC). Nuclear Audit and Review Committee

f. Review of events requiring 24 hour written notification to the Commission.

g. Review of unit operations to detect potential nuclear safety hazards.

Manager

h. Performance of special reviews, investigations ~~or~~ analyses and reports thereon as requested by the *Plant ~~Superintendent)~~ or the ~~(Company Nuclear Review and Audit Group).~~ NARC

i. Review of the Security Plan and implementing procedures and shall submit recommended changes to the ~~(Company Nuclear Review and Audit Group).~~ NARC

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the ~~(Company Nuclear Review and Audit Group).~~ NARC

AUTHORITY

Onsite Review Committee (OSRC)

6.5.1.7 The ~~(Unit Review Group)~~ shall:

Manager

a. Recommend in writing to the *Plant ~~Superintendent)~~ approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.*

OSRC NARC Manager of Nuclear Operations

c. Provide written notification within 24 hours to the ~~A(Superintendent of Power Plants)~~ and the ~~(Company Nuclear Review and Audit Group)~~ of disagreement between the ~~(URG)~~ and the *Plant ~~Superintendent);~~ Manager however, the *Plant ~~Superintendent)~~ shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

Manager

RECORDS

Onsite Review Committee OSRC OSRC

6.5.1.8 The ~~(Unit Review Group)~~ shall maintain written minutes of each ~~(URG)~~ meeting that, at a minimum, document the results of all ~~(URG)~~ activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the ~~(Superintendent of Power Plants)~~ and the ~~(Company Nuclear Review and Audit Group).~~

*(SEE INSERT "A" ATTACHED) NARC. Manager of Nuclear Operations

INSERT "A" (FOOTNOTE)

*"Unreviewed Safety Question", as defined
in 10 CFR 50.59, Rev. January 1, 1980

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SAN ONOFRE UNIT -2 6-7a

NUCLEAR AUDIT AND REVIEW COMMITTEE
6.5.2 ~~COMPANY NUCLEAR REVIEW AND AUDIT GROUP (CNRAG)~~

FUNCTION

Nuclear Audit and Review Committee
6.5.2.1 The ~~(Company Nuclear Review and Audit Group)~~ shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

i. (other appropriate fields associated with the unique characteristics of the nuclear power plant)

COMPOSITION

NARC
6.5.2.2 The ~~(CNRAG)~~ shall be composed of the: [SEE INSERT "A", ATTACHED]

~~Director: (Position Title)~~
~~Member: (Position Title)~~
~~Member: (Position Title)~~
~~Member: (Position Title)~~
~~Member: (Position Title)~~

ALTERNATES

NARC
6.5.2.3 All alternate members shall be appointed in writing by the ~~(CNRAG)~~ Director to serve on a temporary basis; however, no more than two alternates shall participate as voting members in ~~(CNRAG)~~ activities at any one time.
NARC

CONSULTANTS

NARC
6.5.2.4 Consultants shall be utilized as determined by the ~~(CNRAG)~~ Director to provide expert advice to the ~~(CNRAG)~~.
NARC

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INSERT "A"

Manager of Engineering Design
Manager of Environmental Affairs
Manager of Nuclear Engineering, Safety & Licensing
Manager, Quality Assurance
Manager of Nuclear Operations
Manager, Nuclear Engineering & Safety
Manager, Biological Systems Research and Development
San Diego Gas & Electric Representative

Chairmanship shall be designated by the Nuclear Control Board."

MEETING FREQUENCY

6.5.2.5 The NARC ~~(CNRAG)~~ shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 The minimum quorum of the NARC ~~(CNRAG)~~ necessary for the performance of the NARC ~~(CNRAG)~~ review and audit functions of these technical specifications shall consist of the Director or his designated alternate and (at least 4 ~~CNRAG)~~ NARC 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 NARC ~~(CNRAG)~~ shall review:

a. The safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the Onsite Review Committee. ~~(Unit Review Group).~~

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the ~~(CNRAG).~~ NARC These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Emergency Plan and implementing procedures at least once per 24 months.

f. The Security Plan and implementing procedures at least once per 24 months.

g. Any other area of unit operation considered appropriate by the ~~(CNRAG)~~ NARC or the ~~(Vice President Operations).~~ Nuclear Control Board (NCB)

h. The Fire Protection Program and implementing procedures at least once per 24 months.

i. An independent fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or an outside fire protection firm.

j. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

AUTHORITY

6.5.2.9 The ~~(CNRAG)~~ NARC shall report to and advise the ~~(Vice President Operations)~~ NCB on those areas of responsibility specified in Sections 6.5.2.7 and 6.5.2.8.

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(SEE INSERT "A", ATTACHED)

RECORDS

6.5.2.10 Records of NARC ~~(CNRAG)~~ activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NARC ~~(CNRAG)~~ meeting shall be prepared, approved and forwarded to the NCB ~~(Vice President-Operations)~~ within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.7 above, NCB shall be prepared, approved and forwarded to the ~~(Vice President-Operations)~~ within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the NCB ~~(Vice President-Operations)~~ and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the OSRC ~~(URG)~~ and submitted to the NARC ~~(CNRAG)~~ and the Manager of Nuclear Operations ~~(Superintendent of Power plants)~~.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within one hour. The Manager of Nuclear Operations ~~(Superintendent of Power Plants)~~ and the NARC ~~(CNRAG)~~ shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the OSRC ~~(URG)~~. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the NARC ~~(CNRAG)~~ and the Manager of Nuclear Operations ~~(Superintendent of Power Plants)~~ within 14 days of the violation.

INSERT "A"

6.5.3 Nuclear Control Board (NCB)

6.5.3.1 Function

The NCB provides for the corporate level control of the nuclear safety and environmental impact of the facility.

6.5.3.2 Composition

The NCB shall comprise:

- o Vice-President (Power Supply)
- o Vice-President (Quality Assurance)
- o Vice-President (Engineering & Construction)
- o Vice-President (San Diego Gas & Electric Company)

6.5.3.3 Meeting Frequency

The NCB shall meet at least once per 6 months.

6.5.3.4 Quorum

A quorum of the NCB shall consist of two members.

6.5.3.5 Responsibility

The NCB shall:

a. Formally submit safety analysis report to the NRC if a technical specification is violated.

b. Review and approve recommended changes to the technical specifications.

c. Submit proposed changes to the technical specifications to the NRC.

d. Maintain management control with respect to nuclear safety and environmental impact.

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6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

OSRC Manager

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the ~~(URG)~~ and approved by the Plant ~~Superintendent)~~ prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

OSRC

c. The change is documented, reviewed by the ~~(URG)~~ and approved by the Plant ~~Superintendent)~~ within 14 days of implementation.

Manager

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Director of the Regional Office of Inspection and Enforcement unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

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6.9.1.2 The startup report shall address each of the tests following fuel loading, identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated manrem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

[2/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

~~b. The results of the core barrel movement monitoring activities performed during the report period. (CE units only).~~

~~c. (Any other unit unique reports required on an annual basis.)~~

MONTHLY OPERATING REPORT

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the ~~PORVs or~~ safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Director of the Regional Office, or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Failure of the reactor protection system or other systems subject to limiting safety system settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the limiting condition for operation established in the technical specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

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d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the technical specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the SAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the SAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by technical specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the technical specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or technical specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or technical specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

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a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

SPECIAL REPORTS

~~Special reports may be required covering inspections, test and maintenance activities. These special reports are determined on an individual basis for each unit and their preparation and submittal are designated in the Technical Specifications.~~

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE OCCURRENCES submitted to the Commission.

d. Records of surveillance activities, inspections and calibrations required by these Technical Specifications.

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e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the Unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c. Records of radiation exposure for all individuals entering radiation control areas.

d. Records of gaseous and liquid radioactive material released to the environs.

e. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

f. Records of reactor tests and experiments.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the OSRC ~~(URG)~~ and the NARC ~~(CNRAG)~~.

l. Records of the service lives of all ~~hy~~draulic and mechanical snubbers listed on Tables 3.7-4a and 3.7-4b, herein, including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA (OPTIONAL)

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation ~~Work~~ Exposure Permit*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

(REP)

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. An individual qualified in radiation protection procedures who is equipped with a radiation dose rate monitoring device. This individual shall be responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the Radiation Work Permit.

6.12.2 The requirements of 6.12.1, above, shall also apply to each high radiation area in which the intensity of radiation is greater than 1000 mrem/hr. In addition, locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the ~~Shift Foreman on duty and/or the Plant Health Physicist.~~ Watch Engineer

REP

*Health Physics personnel on personnel escorted by Health Physics personnel shall be exempt from the ~~RWP~~ issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

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