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no PORV's

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3/4.6 CONTAINMENT SYSTEMS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

no such systems

not req'd to protect any safety-related area

not req'd due to location, climate, and qualifications

not reqd due to layout of building

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No such system

not req'd at SONGS

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not req'd

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ADMINISTRATIVE CONTROLS

1.0 DEFINITIONS DRAFT

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

ACTION

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

AXIAL SHAPE INDEX

1.2 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.

AZIMUTHAL POWER TILT - T_q

1.3 AXIMUTHAL POWER TILT shall be the power asymmetry between azimuthally symmetric fuel assemblies.

CHANNEL CALIBRATION

1.4 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

1.5 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.

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CHANNEL FUNCTIONAL TEST

1.6 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.

CONTAINMENT INTEGRITY

1.7 CONTAINMENT INTEGRITY shall exist when:

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

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

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

b. All equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

SONGS 2/3 RCPs do not use seal injection

(Not applicable.)

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

CORE ALTERATION

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

DRAFT

DOSE EQUIVALENT I-131

1.10 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."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

for EDG, add 10 sec

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

not applicable

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.

DRAFT

OFFSITE DOSE CALCULATION MANUAL (ODCM)

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

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

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

PRESSURE BOUNDARY LEAKAGE

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

PROCESS CONTROL PROGRAM (PCP)

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

PURGE - PURGING

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

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RATED THERMAL POWER

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

REACTOR TRIP SYSTEM RESPONSE TIME

1.24 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.

REPORTABLE OCCURRENCE

1.25 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Specifications 6.9.1.~~12~~ 8 and 6.9.1.~~13~~ 9.

SHUTDOWN MARGIN

1.26 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming (1) 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, and (2) no change in part length control element assembly position.

SOFTWARE

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

SOLIDIFICATION

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

SOURCE CHECK

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

DRAFT

STAGGERED TEST BASIS

1.30 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

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

VENTILATION EXHAUST TREATMENT SYSTEM

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

VENTING

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

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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%	$\geq$ 350°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 350°F
3. HOT STANDBY	< 0.99	0	$\geq$ 350°F
4. HOT SHUTDOWN	< 0.99	0	350°F> T_{avg}>200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

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

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TABLE 1.2

FREQUENCY NOTATION

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

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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 1.19.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.19, 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.

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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.

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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	(later)	(later)
a. Four Reactor Coolant Pumps Operating	≤ ~~120%~~ of RATED THERMAL POWER	≤ 121.3 ~~123.3%~~ 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)	≤ 0.89% of RATED THERMAL POWER	≤ 0.96% of RATED THERMAL POWER
4. Pressurizer Pressure - High	≤ 2382 psia	≤ 2389 psia
5. Pressurizer Pressure - Low (2)	≥ 1806 psia ~~(2)~~	≥ 1763 psia ~~(2)~~
6. Containment Pressure - High	≤ 2.95 ~~17.65 psia~~ psig	≤ 3.14 ~~17.84 psia~~ psig
7. Steam Generator Pressure - Low (3)	≥ 729 psia ~~(3)~~	≥ 711 psia ~~(3)~~
8. Steam Generator Level - Low (4)	≥ 23% ~~(4)~~	≥ 22.23% ~~(4)~~

being reanalyzed to address a specific high energy line break

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

SAN ONOFRE-UNIT 2

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

sheared shaft

equipment protective trips

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High (5)	≤ 21 kw/ft ~~(5)~~	≤ 21 kw/ft ~~(5)~~
10. DNBR - Low (5)	≥ 1.19 ~~(5)~~	≥ 1.19 ~~(5)~~
11. Reactor Coolant Flow-Low (6)	≥ (later)	≥ (later)
12. ~~11.~~ Steam Generator Level - High (4)	≤ 90 ~~(4)~~	≤ 90.74% ~~(4)~~
13. Seismic-High (7)	≤ .333/.166	≤ .400/.200
14. Loss of Load	Not Applicable	Not Applicable

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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 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 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~~ variable 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.

(6) Rate of decrease of Steam Generator primary side ΔP, psid/sec.

(7) Zero-period acceleration, horizontal/vertical, g's.

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BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

[redacted]

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NOTE

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

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2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The DNBR of the reactor core shall be maintained greater than or equal to 1.19.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the DNBR of the reactor has decreased to less than 1.19, 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.

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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.

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BASES

Linear Power Level-High

(later)

~~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 121.3% of RATED THERMAL POWER.~~

X

Basis [illegible] to [illegible] a portion of [illegible] event

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 0.96% 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 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.

Correction

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 ~~1763~~ 1806 psia. This trip's setpoint may be manually decreased, to a minimum value of 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 400 psi; this setpoint increases automatically as pressurizer pressure increases until the trip setpoint is reached. ||

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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.

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 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 1.19 such that the decrease in actual core

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

Sheared shaft

Reactor Coolant Flow - Low

[see attached]

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 enhances the overall reliability of the Reactor Protection System.

equipment protective trips

Seismic - High

[see attached]

Loss of Load

[see attached]

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Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event. A trip is initiated when the pressure differential accross the primary side of either steam generator decreases at a rate greater than the maximum for coastdown of an intact reactor coolant pump. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under sheared shaft conditions.

Seismic - High

The Seismic - High trip is provided to trip the reactor in the event of an earthquake which exceeds the operating basis earthquake level. 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.

Loss of Load

The Loss of Load trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip is an equipment protective trip only and is not required for plant safety.

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SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

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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.

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

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

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

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.

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, within 2 hours action shall be initiated to place the unit in a MODE in which the applicable Limiting Condition for Operation does not apply by placing it as applicable in:

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

This specification is not applicable in MODE 5 or 6.

APPLICABILITY

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SURVEILLANCE REQUIREMENTS

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

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

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

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

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

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

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

a. Inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing 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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4.0.5 (Continued)

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
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 5.15% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 5.15% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 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 5.15% 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 1 or 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.

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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.

REACTIVITY CONTROL SYSTEMS

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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% delta k/k.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN less than 2.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

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

a. Within one hour after detection of an inoperable 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).

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

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MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be: C-E

a. Less positive than ~~0.5~~ +0.13 x 10^{-4} delta k/k/°F, and

b. Less negative than ~~-3.3~~ -2.5 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 of reaching 40 EFPDs core burnup ~~after the RATED THERMAL POWER equilibrium boron concentration decreases to 900 ppm.~~

c. At any THERMAL POWER, within 7 EFPD of reaching 2/3 of expected core burnup ~~after the RATED THERMAL POWER equilibrium boron concentration decreases to 300 ppm.~~

Boron concentration never expected to reach 900 ppm

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

#See Special Test Exception 3.10.2.

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~~[illegible]~~

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REACTIVITY CONTROL SYSTEMS

MINIMUM TEMPERATURE FOR CRITICALITY

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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 ~~520°F~~ 520 when the reactor is critical. C-E

APPLICABILITY: MODES 1 and 2#.

ACTION:

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

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.

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

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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 Reactor Coolant System if only a 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 storage 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~~ 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.

REACTIVITY CONTROL SYSTEMS

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

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by verifying that the temperature of the heat traced portion of the flow path from the boric acid 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.

While proceeding to or in COLD SHUTDOWN if not performed in the previous 12 m

d. ~~At least once per 18 months when the Reactor Coolant System is at normal operating pressure~~ by verifying that the flow path required be Specification 3.1.2.2.a delivers at least 40 gpm to the Reactor Coolant System.

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

CHARGING PUMPS - OPERATING

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LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2% 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 No additional Surveillance Requirements other than those required by Specification 4.0.5.

BORIC ACID MAKEUP PUMP - SHUTDOWN

DRAFT

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

BORIC ACID MAKEUP PUMPS - OPERATING

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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 2% 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 No additional Surveillance Requirements other than those required by Specification 4.0.5.

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REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCE - SHUTDOWN

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LIMITING CONDITION FOR OPERATION

need for more piping than tank

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 storage tanks with:

1. A minimum contained borated water volume of 5465 gallons, revise to include unrecoverable volume, whichever.

2. A minimum boron concentration of 1720 ppm, and

3. A minimum solution temperature of ~~35~~ 40°F.

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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:

a. At least once per 7 days by:

1. Verifying the boron concentration of the water,

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

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 ~~40~~ ~~35~~ 40°F.

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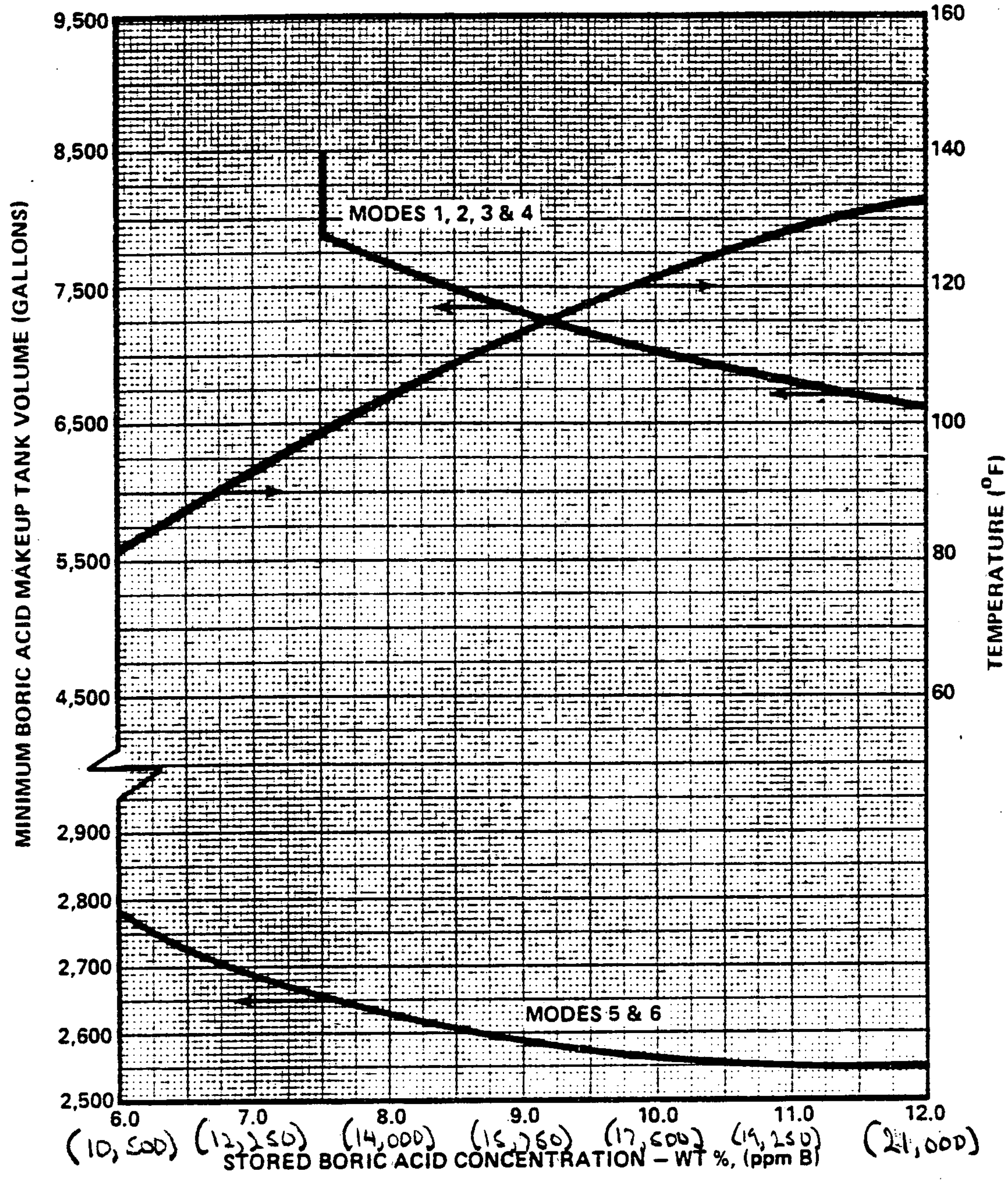

Figure 3.1-1

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

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BORATED WATER SOURCES - OPERATING

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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 with the contents of the tanks in accordance with Figure 3.1-1, and X

b. The refueling water storage tank with:

1. A ~~minimum~~ contained borated water volume of between (later) and (later) ~~355,000 gallons,~~ X

2. Between 1720 and 2300 ppm of boron, ~~and~~ and

3. A minimum solution temperature of ~~35~~ 40°F. X

Same as ANO-2

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 2% 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:

a. At least once per 7 days by:

1. Verifying the boron concentration in the water,

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

3. Verifying the boric acid 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~~ 40°F. X

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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.

See Insert 'A'

b. ~~With one full length CEA inoperable due to causes other than addressed by ACTION a., above, 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, operation in MODES 1 and 2 may continue, if:~~

~~1. The inoperable CEA is in any shutdown or regulating group except group ____ and is fully withdrawn, or~~

~~2. The inoperable CEA is in group ____ and is within the Long Term Steady State Insertion Limits.~~

c ~~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 either:

1. The misaligned CEA(s) is restored to OPERABLE status within its above specified alignment requirements, or

2. 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, or

*See Special Test Exceptions 3.10.2 and 3.10.4.

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INSERT "A"

"With one full length CEA inoperable due to causes other than addressed in Action Item A, operation in Modes 1 and 2 may continue provided:

1. The inoperable CEA is within the specified alignment requirements; and

2. The inoperable CEA is fully withdrawn if in a Shutdown or Regulating group, except Regulating Group 6; or

3. The inoperable CEA is within the Long Term Steady State Inserted Limits if in Regulating Group 6."

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ACTION: (Continued)

3. The misaligned CEA(s) is 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 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.

d ~~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. The misaligned CEA is restored to OPERABLE status within its above specified alignment requirements, or

2. 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, or

3. The misaligned CEA is 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 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 ~~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 length CEAs in its group.

f ~~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.

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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.

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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.

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POSITION INDICATOR CHANNEL - 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:
(85% insertion position for CEAs # 88, 89, 90, & 91)

a. T_{avg} greater than or equal to 520°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.

SURVEILLANCE REQUIREMENTS

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

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.

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REACTIVITY CONTROL SYSTEMS

SHUTDOWN CEA INSERTION LIMIT

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LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to within 3.0 inches of the Full Out position.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 3.0 inches of 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 within 3.0 inches of the Full Out position, or

b. Declare the CEA inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each shutdown CEA shall be determined to be withdrawn to within 3.0 inches of 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

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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,

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

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

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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:
(effective full power)

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

4.1.3.5 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. X

to be within the Transient Insertion Limits (and within 3 inches of the full-out position in a fully withdrawn regulating CEA group)

since wear cycling is per CEA, not per group

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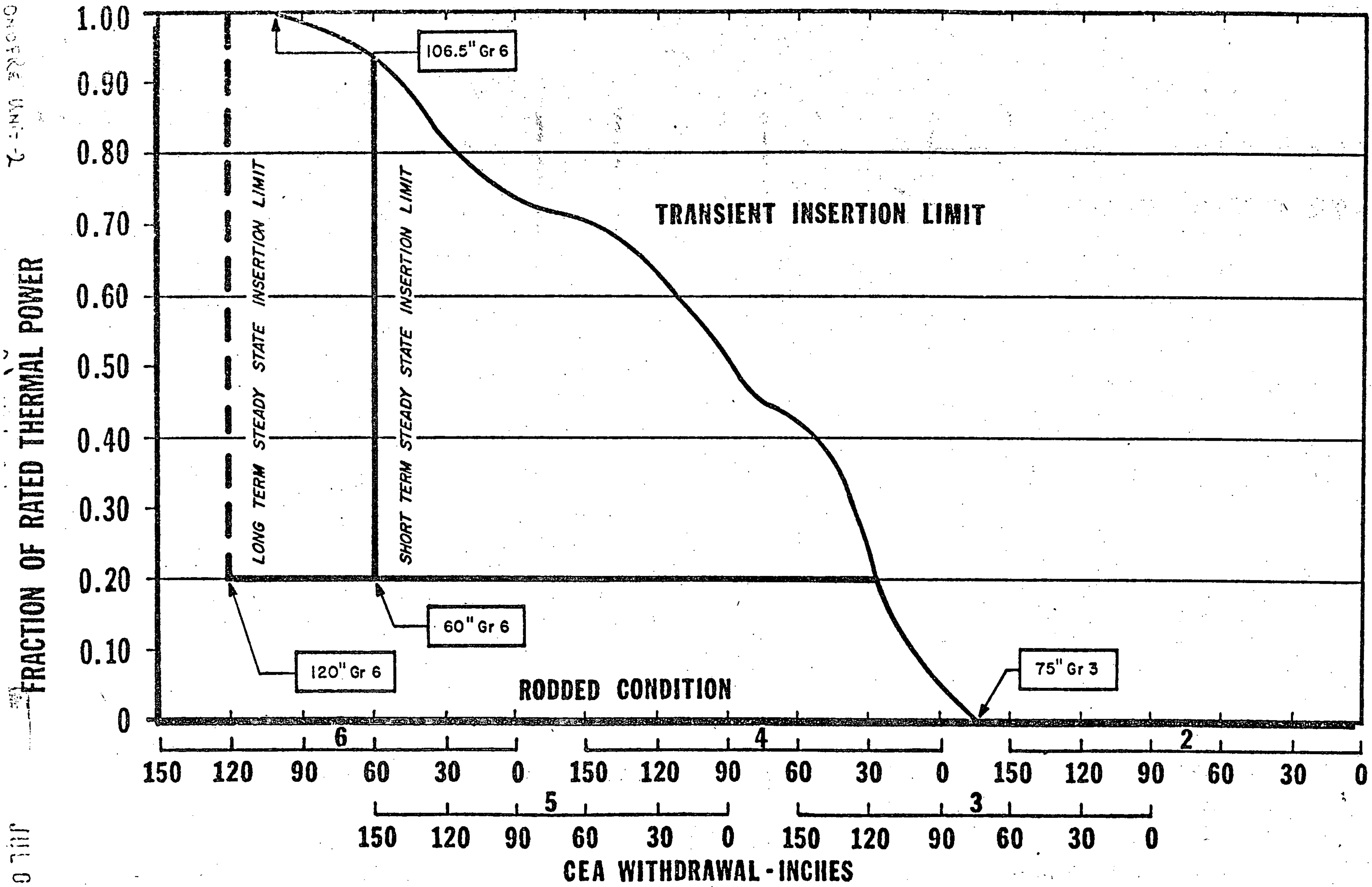

Figure 3.1-2

CEA ISERTION LIMITS vs THERMAL POWER

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PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.7 The position of the Inconel 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 Inconel 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.

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3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 LINEAR HEAT RATE

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LIMITING CONDITION FOR OPERATION

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

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, when COLSS is in service 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.

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Figure 3.2-1

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CE to provide/verify

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3/4.2.2 RADIAL PEAKING FACTORS - Fr

LIMITING CONDITION FOR OPERATION

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

when COLSS is being used to monitor the core [illegible] per Spec [illegible] and 4.2.4.

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

ACTION:

With a F_{xy}^m exceeding a corresponding F_{xy}^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_{xy}^m/F_{xy}^c restrict subsequent operation so that a margin to the COLSS operating limits of at least $[(F_{xy}^m/F_{xy}^c) - 1.0] \times 100\%$ is maintained; or

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

c. Be in at least HOT STANDBY.

NOTE: Fr changed To Fxy for Consistency with C-E Terminology

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured planar radial peaking factors (F_{xy}^m) obtained by using the incore detection system, shall be determined to be less than or equal to the planar radial peaking factors (F_{xy}^c), used in the COLSS (when COLSS is being used) 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.

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3/4.2.3 AZIMUTHAL POWER TILT - Tq

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LIMITING CONDITION FOR OPERATION

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

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

ACTION:

LATER CE to verify

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; sbsequent 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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4.2.3 The AZIMUTHAL POWER TILT shall be determined to be within the limit above {20}% of RATED THERMAL POWER by: either

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.

b, ~~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.

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

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3/4.2.4 DNBR MARGIN

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LIMITING CONDITION FOR OPERATION

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.

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SURVEILLANCE REQUIREMENTS (Continued)

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 (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]

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Figure 3.2-3

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POWER DISTRIBUTION LIMITS

3/4.2.5 RCS FLOW RATE

LIMITING CONDITION FOR OPERATION

3.2.5 The actual Reactor Coolant System total flow rate shall be greater than or equal to ~~(120.4 x 10⁶)~~ 148 x 10⁶ lbm/hr. and less than or equal to 177.6x10⁶ lbm/hr. x

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

3.2.6 The core average coolant temperature (T_{avg}) shall be less than or equal to ~~(588.2)~~ 588.05 °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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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#
3. Logarithmic Power Level-High					
a. Startup and Operating	4	2(a)(d)	3	1, 2, and *	2#, 3#
b. Shutdown	4	0	2	3, 4, 5	4
4. Pressurizer Pressure - High	4	2	3	1, 2	2#, 3#
5. Pressurizer Pressure - Low	4	2(b)	3	1, 2	2#, 3#
6. Containment Pressure - High	4	2	3	1, 2	2#, 3#
7. Steam Generator Pressure - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
9. Local Power Density - High	4	2(c)(d)	3	1, 2	2#, 3#
10. DNBR - Low	4	2(c)(d)	3	1, 2	2#, 3#
11. Steam Generator Level - High	4/SG	2/SG	3/SG	1, 2	2#, 3#
12. Reactor Protection System Logic	~~2~~ 4	~~1~~ 2	~~2~~ 3	1, 2, and *	~~5~~ 2#, 3#
13. Reactor Trip Breakers	2	1 (f)	2	1, 2, and *	5
14. Core Protection Calculators	4	2(c)(d)	3	1, 2	2#, 3# and 7
15. CEA Calculators	2	1	2(e)	1, 2	6 and 7
16. Reactor Coolant Flow - Low	4/SG	2/SG	3/SG	1, 2	2#, 3#
17. Seismic - High	4	2	(g)	(g)	(g)
18. Loss of Load	4	2	(g)	(g)	(g)

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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~~ 400 psia. (per FSAR Table 7.2-1)

(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.

(g) Equipment protective trip only; not required for reactor protection

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.

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 inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the channel shall be returned to OPERABLE status within 90 days for failures outside containment and within 18 months for failures inside containment, and the bypass removed.

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip 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

ACTION 3 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other channel in the tripped condition within 1 hour, and

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

Process Measurement Circuit	Functional Unit Bypassed/Tripped
1. Linear Power (Subchannel or Linear)	Linear Power Level - High Local Power Density - High DNBR - Low

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TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

The bypassed channels shall be returned to OPERABLE status within 90 days of the date of bypass for failures outside containment and within 18 months of the date of bypass for failures inside containment, and the bypass removed.

ACTION 4 - 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 one channel to OPERABLE status ~~and~~ or provide an alternate source level monitoring.

ACTION 5 - 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 6 - 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

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TABLE 3.3-1 (Continued)

TABLE NOTATION

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 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 7 - 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	≤ 0.40 seconds*
3. Logarithmic Power Level - High	≤ 0.45 seconds*
4. Pressurizer Pressure - High	≤ ~~1.50~~ .90 seconds
5. Pressurizer Pressure - Low	≤ 0.90 seconds
6. Containment Pressure - High	≤ 0.90 seconds
7. Steam Generator Pressure - Low	≤ 0.90 seconds
8. Steam Generator Level - Low	≤ 0.90 seconds
9. Local Power Density - High	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ____ seconds*
b. CEA Positions	≤ ____ seconds**
c. CEA Positions as transmitted by CEAC Penalty Factor	≤ ____ seconds
10. DNBR - Low	
a. Neutron Flux Power from Excore Neutron Detectors	≤ ____ seconds*
b. CEA Positions	≤ ____ seconds**
c. Cold Leg Temperature	≤ ____ seconds##
d. Hot Leg Temperature	≤ ____ seconds##
e. Primary Coolant Pump Shaft Speed	≤ ____ seconds#
f. Reactor Coolant Pressure from Pressurizer	≤ ____ seconds
g. CEA Positions as transmitted by CEAC Penalty Factor	≤ ____ seconds*##

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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
16. Reactor Coolant Flow - Low	(Later)
17. Seismic - High	Not Applicable
18. Loss of Load	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.

**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.

###Based on a pressure transmitter response time of less than or equal to 0.7 seconds where the pressure transmitter response time is equivalent to the time interval required for the transmitter to achieve 63.2% of its total change when subjected to a step change in pressure transmitter pressure.

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TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	1, 2, and *
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 *

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TABLE 4.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS 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
16. Reactor Coolant Flow - Low	S	R	M	1, 2
17. Seismic - High	S	R	M	1,2 and *
18. Loss of Load	S	N.A.	M	1 (9)

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TABLE NOTATION

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

(1) - Each startup or when required with the reactor trip breakers closed and the CEA drive system capable of rod withdrawal, if not performed in the previous 7 days. R1

(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).

(9) - Above 55% of RATED THERMAL POWER.

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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	8
b. Containment Pressure - High	4	2	3	1, 2, 3	9*, 10*
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	9*, 10*
d. Automatic Actuation - Logic	2	1	2	1, 2, 3	12
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	9*, 10*
c. Automatic Actuation Logic	2	1	2	1, 2, 3	12
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Manual SIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
c. Containment Pressure - High	4	2	3	1, 2, 3	9*, 10*
d. Automatic Actuation Logic	2	1	2	1, 2, 3	12

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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
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3~~(c)~~	9*, 10* ?
c. Automatic Actuation Logic	2/steam generator	1/steam generator	2/steam generator	1, 2, 3	12
5. RECIRCULATION (RAS)					
a. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	9*, 10*
b. Automatic Actuation Logic	2	1	2	1, 2, 3	12
6. CONTAINMENT COOLING (CCAS)					
a. Manual (Trip Buttons)	2 ~~sets of 2~~	1 ~~set of 2~~	2 ~~sets of 2~~	1, 2, 3, 4	8 ?
b. Containment Pressure - High	4	2	3	1, 2, 3,	9, 10
c. Pressurizer Pressure - Low	4	2	3	1, 2, 3(a)	9, 10
d. Automatic Actuation Logic	2	1	2	1, 2, 3	12

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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. LOSS OF POWER					
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded Voltage)	4/Bus	2/Bus	3/Bus	1, 2, 3, 4	~~8~~ 9*, 10*
~~b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)~~	~~4/Bus~~	~~2/Bus~~	~~3/Bus~~	~~1, 2, 3, 4~~	~~8~~
8. 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
b. Automatic Actuation Logic	2/SG	1/SG	2/SG	1, 2, 3	12
c. SG Level and Pressure (A/B) - Low and ΔP (A/B) - High	4/SG	2/SG	3/SG	1, 2, 3	9*, 10*
d. SG Level (A/B) - Low and No S/G Pressure - Low Trip (A/B)	4/SG	2/SG	3/SG	1, 2, 3	9*, 10*

Same set of devices performs both functions

Variable Trip Pt

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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. Control Room Isolation (CRIS)					
a. Manual CRIS (Trip Buttons)	2	1	1	All	13 *#
b. Manual SIAS (Trip Buttons)	2/unit	1	2/unit	1,2,3,4	8
c. Airborn Radiation					
i. Particulate/Iodine	2	1	1	All	13 *#
ii. Gaseous	2	1	1	All	13 *#
d. Automatic Actuation Logic	2	1	1	All	13 *#
10. Toxic Gas Isolation (TGIS)					
a. Manual (Trip Buttons)	2	1	1	All	14*#,15*#
b. Chlorine - High	2	1	1	All	14*#,15*#
c. Ammonia - High	2	1	1	All	14*#,15*#
d. Butane/Propane - High	2	1	1	All	14*#,15*#
e. Carbon Dioxide - High	2	1	1	All	14*#,15*#
f. Automatic Actuation Logic	2	1	1	All	14*#,15*#

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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
11. Fuel Handling Isolation (FHIS)					
a. Manual (Trip Buttons)	2	1	1	**	16 *#
b. Airborne Radiation					
i. Gaseous	2	1	1	**	16 *#
ii. Particulate/Iodine	2	1	1	**	16 *#
c. Automatic Actuation Logic	2	1	1	**	16 *#
12. Containment Purge Isolation (CPIS)					
a. Manual (Trip Buttons)	2	1	1	6	17 *#
b. Airborne Radiation					
i. Gaseous	2	1	1	6	17 *#
ii. Particulate	2	1	1	6	17 *#
iii. Iodine	2	1	1	6	17 *#
c. Containment Area Radiation (Gamma)	2	1	1	6	17 *#
d. Automatic Actuation Logic	2	1	1	6	17 *#

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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)~~ 400 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.~~ Delete variable S.P.

\# The provisions of Specification 3.0.3 are not applicable.

* The provisions of Specification 3.0.4 are not applicable.

** With irradiated fuel in the storage pool.

ACTION STATEMENTS

ACTION 8 - 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 9 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the channel shall be returned to OPERABLE status within 90 days for failures outside containment and within 18 months for failures inside containment, and the bypass removed.

With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip 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)
3.	Steam Generator Level - Low	Steam Generator Level - Low Steam Generator Level - High Steam Generator ΔP (EFAS)

TABLE 3.3-3 (Continued)

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TABLE NOTATION

ACTION 10 - With the number of channels OPERABLE one less than the minimum channels OPERABLE, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. Verify that one of the inoperable channels has been bypassed and place the other inoperable channel in the tripped condition within 1 hour.

b. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition as listed below:

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

The bypassed channel shall be returned to OPERABLE status within 90 days of the date of bypass for failures outside containment and within 18 months of the date of bypass for failures inside containment, and the bypass removed.

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.

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TABLE 3.3-3 ACTION STATEMENTS (continued)

Action 13	With the number of channels OPERABLE less than required by the mimimum channels OPERABLE requirement, within 1 hour initiate and maintain operation of the control room emergency air cleanup system in the emergency (except as required by ACTIONS 14,15) mode of operation.
Action 14	With the number of channels OPERABLE one less than the total number of channels, restore the inoperable channel to OPERABLE status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency air cleanup system in the isolation mode of operation.
Action 15	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 air cleanup system in the isolation mode of operation.
Action 16	With the number of channels OPERABLE less than required by the minimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.11.
Action 17	With the number of channels OPERABLE less than required by the mimimum channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.8.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)	2.75	3.14
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ ~~17.45 psia~~ 2.95 psig	≤ ~~17.84 psia~~ 3.14 psig
c. Pressurizer Pressure - Low	≥ 1806 psia (1)	≥ 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	≤ ~~22.84 psia~~ 8.14 psig	≤ ~~23.43 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	≤ ~~17.45 psia~~ 2.95 psig	≤ ~~17.84 psia~~ ~~2.95 psig~~ 3.14 psig
d. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 729 psia (2)	≥ 711 psia (2)
c. Automatic Actuation Logic	Not Applicable	Not Applicable
5. RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank	~~18%~~ 18.5% of tap span	19.26% ≥ tap span ≥ 17.74%
c. Automatic Actuation Logic	Not Applicable	Not Applicable

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TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

corrected

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ ~~17.45 psia~~ 2.95 psig	≤ ~~17.84 psia~~ 3.14 psig
c. Pressurizer Pressure - Low	≥ 1806 psia (1)	≥ 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	≤ ~~22.84 psia~~ ~~8.14 psig~~ (later)	≤ ~~23.43 psia~~ ~~8.22 psig~~ (later)
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	≤ ~~17.45 psia~~ 2.95 psig	≤ ~~17.84 psia~~ ~~2.95 psig~~ 3.14 psig
d. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 729 psia (2)	≥ 711 psia (2)
c. Automatic Actuation Logic	Not Applicable	Not Applicable
5. RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank	~~18%~~ 18.5% of tap span	19.26% ≥ tap span ≥ 17.74%
c. Automatic Actuation Logic	Not Applicable	Not Applicable

to be revised for analysis for setpoint of 20 psig in lieu of 12 psig

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	< ~~17.45 psia~~ 2.95 psig	< ~~17.84 psia~~ ~~2.95 psig~~ 3.14 psig
c. Pressurizer Pressure - Low	> 1806 psia	> 1763 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable
7. LOSS OF POWER (LOV)	Figure 3.3-1 (4)	Figure 3.3-1 (4)
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage and Degraded 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~~
8. EMERGENCY FEEDWATER (EFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (A&B) Level-Low	> 23% (3)	> 22.23% (3)
c. Steam Generator ΔP-High (SG-A > SG-B)	< 50 psi	< 66.25 psi
d. Steam Generator ΔP-High (SG-B > SG-A)	< 50 psi	< 66.25 psi
e. Steam Generator (A&B) Pressure - Low	> 729 psia (2)	> 711 psia (2)
f. Automatic Actuation Logic	Not Applicable	Not Applicable

same instrument performs both functions

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

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FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. CONTAINMENT COOLING (CCAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	< ~~17.45 psia~~ 2.95 psig	< ~~13.84 psia~~ ~~2.95 psig~~ 3.14 psig
c. Pressurizer Pressure - Low	> 1806 psia	> 1763 psia
d. Automatic Actuation Logic	Not Applicable	Not Applicable
7. LOSS OF POWER (LOV)		
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage~~s~~ and Degraded Voltage)	Figure 3.3-1 (4) ~~(3120) volts~~	Figure 3.3-1 (4) ~~(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~~
8. EMERGENCY FEEDWATER (EFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator (A&B) Level-Low	> 23% (3)	> 22.23% (3)
c. Steam Generator ΔP-High (SG-A > SG-B)	< 50 psi	< 66.25 psi
d. Steam Generator ΔP-High (SG-B > SG-A)	< 50 psi	< 66.25 psi
→ e. Steam Generator (A&B) Pressure - Low	> 729 psia (2)	> 711 psia (2)
f. Automatic Actuation Logic	Not Applicable	Not Applicable

same inst. performs both functions

[INSERT]

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
9. Control Room Isolation (CRIS)		
a. Manual CRIS (Trip Buttons)	Not Applicable	Not Applicable
b. Manual SIAS (Trip Buttons)	Not Applicable	Not Applicable
c. Airborn Radiation		
i. Particulate/Iodine	$\leq$ (later)	$\leq$ ($2x10^{-9}$ μ c/cm^3)
ii. Gaseous	$\leq$ (later)	$\leq$ ($2x10^{-6}$ μ c/cm^3
d. Automatic Actuation Logic	Not Applicable	Not Applicable
10. Toxic Gas Isolation (TGIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Chlorine - High	$\leq$ (later)	$\leq$ (5ppm)
c. Ammonia - High	$\leq$ (later)	$\leq$ (50 ppm)
d. Butane/Propand - High	$\leq$ (later)	$\leq$ (100 ppm)
e. Carbon Dioxide - High	$\leq$ (later)	$\leq$ (5000ppm)
f. Automatic Actuation Logic	Not Applicable	Not Applicable

FSAR analysis set pts

FSAR analysis set pts

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TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
11. Fuel Handling Isolation (FHIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	≤ (later)	≤ (2x10^{-6} μ ci/cm^3)
ii. Particulate/Iodine	≤ (later)	≤ (2x10^{-9} μ ci/cm^3
c. Automatic Actuation Logic	Not Applicable	Not Applicable
12. Containment Purge Isolation (CPIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Airborne Radiation		
i. Gaseous	≤ (later)	≤ (2.7 x 10^{-2}μc/cm^3
ii. Particulate	≤ (later)	≤ (1.8 x 10^{-5}μc/cm^3)
iii. Iodine	≤ (later)	≤ (1.4 x 10^{-5}μc/cm^3)
c. Containment Area Radiation (Gamma)	≤ (later)	≤ (2.5 mr/hr)
d. Automatic Actuation Logic	Not Applicable	Not Applicable

FSAR analysis set pts

Figure 3.3-1

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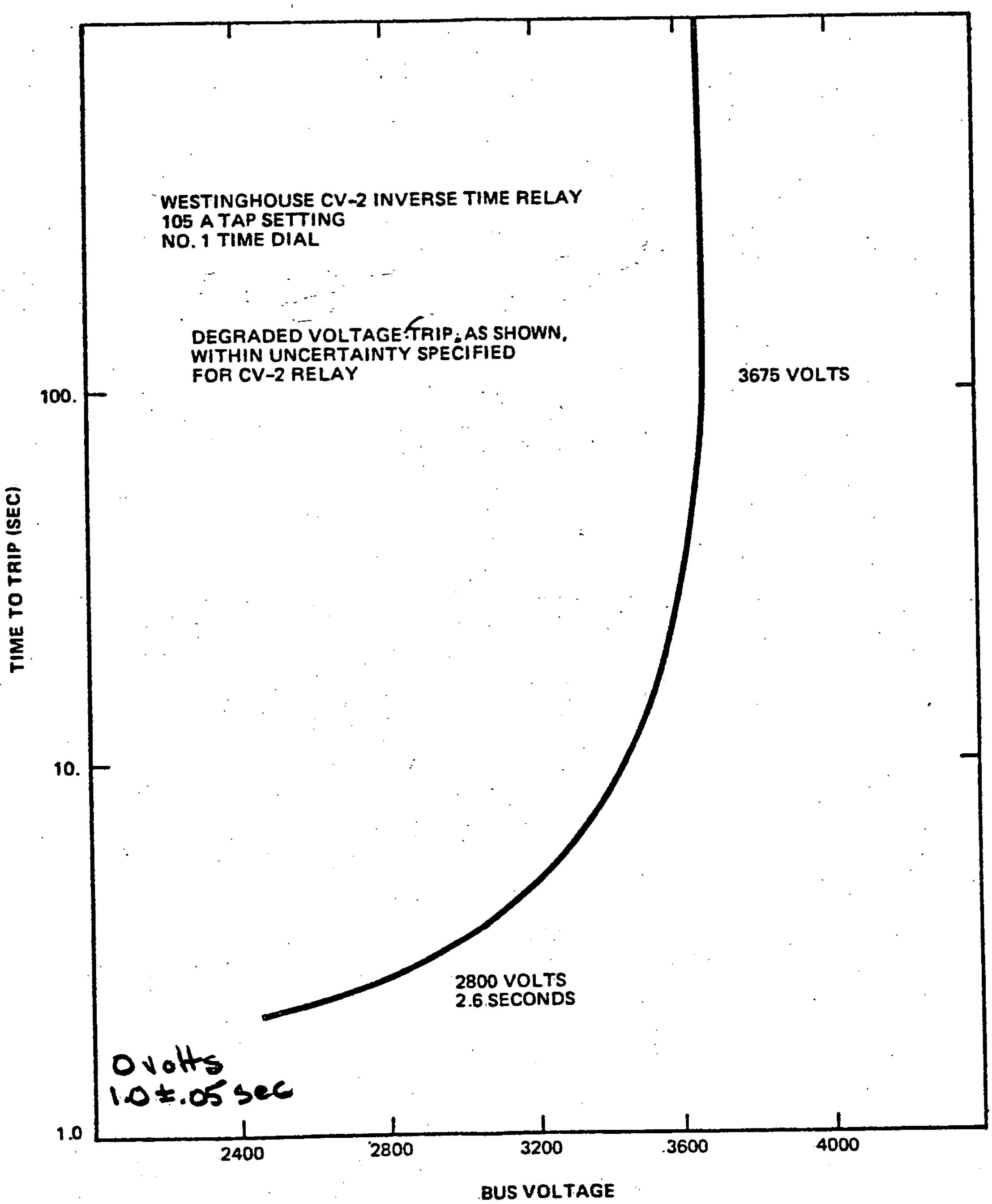

SAN ONOFRE NUCLEAR GENERATING STATION Units 2 & 3
DEGRADED BUS VOLTAGE TRIP SETTING

5/79 15-040.

TABLE 3.3-4 (Continued)

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.

(4) Inverse time relay set value 3165 V; trip will occur within the tolerances specified in Figure 3.3-1 for the range of bus voltages. ×

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

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TABLE 3.3-5

ENGINEERED SAFETY FEATURE RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME
1. Manual	
a. SIAS	
Safety Injection	Not Applicable
Control Room Isolation	Not Applicable
Containment Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIAS	
Containment Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
e. RAS	
Containment Sump Recirculation	Not Applicable
f. CCAS	
Containment Emergency Cooling	Not Applicable
g. EFAS	
Auxiliary Feedwater	Not Applicable
h. CRIS	
Control Room Isolation	Not Applicable
i. TGIS	
Toxic Gas Isolation	Not Applicable
j. FHIS	
Fuel Handling Building Isolation	Not Applicable
k. CPIS	
Containment Purge Isolation	Not Applicable

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Table 3.3-5
Engineered Safety Feature Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME
2. Pressurizer Pressure - Low	
a. SIAS	BPC to verify
1. Safety Injection	
a. High Pressure Safety Injection	(11.2)
b. Low Pressure Safety Injection	(16.2*)
2. Control Room Isolation	Not Applicable
3. Containment Isolation	(31.2)
4. Containment Spray (Pumps)	(15.6*)
5. Containment Emergency Cooling (CCW)	(21.2*)
b. CCAS	
1. Containment Emergency Cooling	(11.2)
3. Containment Pressure - High	
a. SIAS	
1. Safety Injection	
a. High Pressure Safety Injection	(11.0)
b. Low Pressure Safety Injection	(16.0*)
2. Control Room Isolation	Not Applicable
3. Containment Spray (Pumps)	(15.4*)
4. Containment Emergency Cooling (CCW)	(21.0*)
b. CIAS	
1. Containment Isolation	(30.9)
c. CCAS	
1. Containment Emergency Cooling	(11.0)
4. Containment Pressure - High-High	
a. CSAS	
1. Containment Spray	(11.0)

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Table 3.3-5
Engineered Safety Feature Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME
5. Steam Generator Pressure - Low	
a. MSIS	
1. Main Steam Isolation	(5.9) MSIV
2. Main Feedwater Isolation	(10.9)
6. Refueling Water Storage Tank - Low	
a. RAS	
1. Containment Sump Valves Open	(later)
2. ECCS Miniflow Valves Shut	Not Applicable
7. 4.16 kv Emergency Bus Undervoltage	
a. LOV (loss of voltage and degraded voltage)	Figure 3.3-1
8. Steam Generator Level - Low and no Pressure Low Trip	
a. EFAS	valves pumps
1. Auxiliary Feedwater (AC trains)	(30/50*)
2. Auxiliary Feedwater (steam/DC train)	(300)
9. Steam Generator Level - Low and AP - High	
a. EFAS	
1. Auxiliary Feedwater (AC trains	(30/50*)
2. Auxiliary Feedwater (Steam/DC train)	(300)
10. Control Room Ventilation Airborne Radiation	
a. CRIS	
1. Control Room Ventilation - Emergency Mode	Not Applicable
11. Control Room Toxic Gas (Chlorine)	
a. TGIS	isolation dampers only
1. Control Room Ventilation - Isolation Mode	(16)
12. Control Room Toxic Gas (Ammonia)	
a. TGIS	ditto
1. Control Room Ventilation - Isolation Mode	(36)

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Table 3.3-5
Engineered Safety Feature Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME
13. Control Room Toxic Gas (Butane/Propane)	BPC to verify
a. TGIS	
1. Control Room Ventilation - Isolation Mode	(36) ditto
14. Control Room Toxic Gas (Carbon Dioxide)	
a. TGIS	
1. Control Room Ventilation - Isolation Mode	(36) ditto
15. Fuel Handling Building Airborne Radiation	
a. FHIS	
1. Fuel Handling Emergency Air Cleanup	Not Applicable
16. Containment Airborne Radiation	
a. CPIS	
1. Containment Purge Isolation	(13)
17. Containment Area Radiation	
a. CPIS	
1. Containment Purge Isolation	(13)

NOTES:

1. Response times include movement of valves and attainment of pump or blower discharge pressure as applicable.

   * Sequence loading delays for SIAS are included.

2. If "Not Applicable", no response time has been assumed in safety analysis.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS 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. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
4. MAIN STEAM 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. ~~CONTAINMENT SUMP~~ RECIRCULATION (RAS)				
a. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
b. Automatic Actuation Logic	N.A.	N.A.	M(1)	1, 2, 3
6. 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

TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage ~~)~~ and Degraded Voltage)	S	R	R	1, 2, 3, 4
~~b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)~~	~~S~~	~~R~~	~~R~~	~~1, 2, 3~~
8. 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

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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.

TABLE 4.3.-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. Control Room Isolation (CRIS)				
a. Manual CRIS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Manual SIAS (Trip Buttons)	N.A.	N.A.	R	N.A.
c. Airborne Radiation				
i. Particulate/Iodine	S	R	M	All
ii. Gaseous	S	R	M	All
d. Automatic Actuation Logic	N.A.	N.A.	M(1)	All
10. Toxic Gas Isolation (TGIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Chlorine - High	S	R	M	All
c. Ammonia - High	S	R	M	All
d. Butane/Propane - High	S	R	M	All
e. Carbon Dioxide - High	S	R	M	All
f. Automatic Actuation Logic	N.A.	N.A.	M(1)	All

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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 FOR WHICH SURVEILLANCE IS REQUIRED
11. Fuel Handling Isolation (FHIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	*
ii. Particulate/Iodine	S	R	M	*
c. Automatic Actuation Logic	N.A.	N.A.	R	*
12. Containment Purge Isolation (CPIS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Airborne Radiation				
i. Gaseous	S	R	M	6
ii. Particulate	S	R	M	6
iii. Iodine	S	R	M	6
c. Containment Area Radiation (Gamma)	S	R	M	6
d. Automatic Actuation Logic	N.A.	N.A.	R	6

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INSTRUMENTATION

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING ALARM INSTRUMENTATION

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Note: ESFAS trip reqs moved to ESFAS section

LIMITING CONDITION FOR OPERATION

3.3.3.1 The radiation monitoring alarm 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 alarm 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 alarm 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 alarm 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.

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TABLE 3.3-6

RADIATION MONITORING ALARM INSTRUMENTATION

	INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM SETPOINT	MEASUREMENT RANGE	ACTION
	1. Area Monitors					
(**	a. Containment (Area)	2	1,2,3,4	(later) R/hr	$1\text{-}10^8$ R/hr	19)
	b. Containment (Purge Isolation)	1	1,2,3,4: 6 :	≤ 670 MR/hr ≤ 2.5 MR/hr	$10^{-1}\text{-}10^5$ Mr/hr	(a)
(**	c. Main Steam Line	1/line	1,2,3,4	(later) R/hr	$1\text{-}10^3$ R/hr	19)
	2. Process Monitors					
	a. Fuel Storage Pool Airborne					
	i. Gaseous	1	*	≤ 2 x background	10^1 -10^7 cpm	(b)
	ii. Particulate/Iodine	1	*	≤ 2 x background	10^1 -10^7 cpm	(b)
	b. Containment Airborne					
	i. Gaseous	1	1,2,3,4: 6 :	NOT APPLICABLE ≤ 2 x background	10^1 -10^7 cpm	18 (a)
	ii. Particulate	1	1,2,3,4: 6 :	NOT APPLICABLE ≤ 2 x background	10^1 -10^7 cpm	18 (a)
	iii. Iodine	1	1,2,3,4: 6 :	NOT APPLICABLE ≤ 2 x background	$10^1 - 10^7$ cpm	18 (a)
	c. Control Room Airborne					
	i. Particulate/Iodine	1	All	≤ 2 x background	10^1 -10^7 cpm	(c)
	ii. Gaseous	1	All	≤ 2 x background	10^1 -10^7 cpm	(c)

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TABLE 3.3-6 (Continued)

ACTION STATEMENTS

These are now in ESFAS spec

ACTION ~~13~~ 18 - 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~~5.1).

~~ACTION 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).~~

~~ACTION 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).~~

~~ACTION 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.~~

ACTION ~~17~~ 19 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirement, restore the inoperable Channel(s) to OPERABLE status within 7 days, or be in at least HOT STANDBY within the next 6 hours, and in at least HOT SHUTDOWN within the following 6 hours and in COLD SHUTDOWN within the subsequent 24 hours.

TABLE NOTES:

* With irradiated fuel in the storage pool.

** NUREG 0737 item to be operational by January 1, 1982.

(a) In accordance with TABLE 3.3-3 Action 17.

(b) In accordance with TABLE 3.3-3 Action 16.

(c) In accordance with TABLE 3.3-3 Action 13.

[redacted]

TABLE 4.3-3

RADIATION MONITORING ALARM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
	1. Area Monitors				
(**	a. Containment (Area)	S	R	M	1,2,3,4)
	b. Containment (Purge Isolation)	#	#	#	1,2,3,4,& 6
(**	c. Main Steam Line	S	R	M	1,2,3,4)
	2. Process Monitors				
	a. Fuel Storage Pool Airborne				
	i. Gaseous	#	#	#	*
	ii. Particulate/Iodine	#	#	#	*
	b. Containment Airborne				
	i. Gaseous	#	#	#	1,2,3,4 & 6
	ii. Particulate	#	#	#	1,2,3,4 & 6
	iii. Iodine	#	#	#	1,2,3,4 & 6
	c. Control Room Airborne				
	i. Particulate	#	#	#	All
	ii. Gaseous	#	#	#	All

NOTES:
In accordance with table 4.3-2 surveillance requirements for these instrument channels.
* With inradiated fuel in the storage pool.
** NUREG 0737 item to be operational by January 1, 1982.

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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

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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.

greater than or equal to 50% of the Operating Basis Earthquake accelerations

alt: defer calib to next cold S/D

threshhold needed to preclude frequent shutdown just for calibration of sensitive instruments in high radiation areas

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TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

Instruments & Sensor Locations	Measurement Range	Minimum Instrument 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 Reading 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		
6. Peak Shock Recorder		
a. Containment Base in Tendon Gallery	2 to 25.4 Hz 1.6 to 90g	1**
7. Peak Shock Annunciator	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 AND 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	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)

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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.

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TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		
a. 0-50 mph	Nominal Elev. 10 meters	1
b. 0-50 mph	Nominal Elev. 20 meters	1
c. (later)	Nominal Elev. 40 meters	1
2. WIND DIRECTION		
a. 0-360-180°	Nominal Elev. 10 meters	1
b. 0-360-180°	Nominal Elev. 20 meters	1
c. 0-360-180°	Nominal Elev. 40 meters	1
3. AIR TEMPERATURE		
a. -30 to +50°C	Nominal Elev. 10 meters	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

X

new requirements for backup tower promulgated?

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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		
a. Nominal Elev. 10 meters	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

REMOTE SHUTDOWN INSTRUMENTATION

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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

INSTRUMENT	READOUT LOCATION	CHANNELS RANGE	MINIMUM CHANNELS OPERABLE
1. Log Power Level	*	10^{-8} - 200%	1
2. Reactor Coolant Cold Leg Temperature	*	0-600°F	1/loop
3. Pressurizer Pressure	*	0-3000 psia	1
4. Pressurizer Level	*	0-100%	1
5. Steam Generator Pressure	*	0-1200 psia	1/steam generator
6. Steam Generator Level	*	0-100%	1/steam generator
7. Reactor Coolant Boron Concentration	*	0-2500 ppm	1
8. Condenser Vacuum	*	0-5" Hg	1
9. Volume Control Tank Level	*	0-100%	1
10. Letdown Heat Exchanger Pressure	*	0-600 psig	1
11. Letdown Heat Exchanger Temperature	*	0-200°F	1
12. Boric Acid Makeup Tank Level	*	0-100%	1
13. Condensate Storage Tank Level	*	0-100%	1

*Panel L04~~1~~2

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TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Log Power Level	M	R
2. Reactor Coolant Cold Leg Temperature	M	R
3. Pressurizer Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Level	M	R
6. Steam Generator Pressure	M	R
7. Reactor Coolant Boron Concentration	M	R
8. Condenser Vacuum	M	R
9. Volume Control Tank Level	M	R
10. Letdown Heat Exchanger Pressure	M	R
11. Letdown Heat Exchanger Temperature	M	R
12. Boric Acid Makeup Tank Level	M	R
13. Condensate Storage Tank Level	M	R

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INSTRUMENTATION

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ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE accident monitoring channels less than the Required Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 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.

Note:
Post-LOCA H_2: 3/4.6.4
Hi-range Rad: 3/4.3.3

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TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure — Narrow Range	2	1
2. Containment Pressure — Wide Range	2	1
3 ~~2~~. Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	2	1
4 ~~3~~. Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1
5 ~~4~~. Pressurizer Pressure - Wide Range	2	1
6 ~~5~~. Pressurizer Water Level	2	1
7 ~~6~~. Steam Line Pressure	2/steam generator	1/steam generator
8 ~~7~~. Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
9 ~~8~~. Steam Generator Water Level - Wide Range	2 ~~1~~/steam generator	(a) ~~1/steam generator~~
10 ~~9~~. Refueling Water Storage Tank Water Level	2	1
11 ~~10~~. Auxiliary Feedwater Flow Rate	~~2~~ 1/steam generator	(a) ~~1~~
12 ~~11~~. Reactor ~~Cooling~~ Coolant System Subcooling Margin Monitor	2	1
~~12. PORV Position Indicator~~	~~2/valve~~	~~1/valve~~
~~13. PORV Block Valve Position Indicator~~	~~2/valve~~	~~1/valve~~
13 ~~14~~. Safety Valve Position Indicator	2/valve	1/valve
14 ~~15~~. Spray System Temperature	2	1
15 ~~16~~. Spray System Pressure	2	1
16 ~~17~~. LPSI Header Temperature	2	1
17 ~~18~~. Containment Temperature	2	1
18 ~~19~~. Containment Water Level - Narrow Range	2	1
19 ~~20~~. Containment Water Level - Wide Range	2	1
20 ~~21~~. ~~Incore~~ Thermocouples Core Exit	~~(8)~~/core quadrant 7	~~(4)~~/core quadrant 4

No PORV's

Core exit tc's used for ICC at SO2/3

per RG1.97, aux fw flow and S/G wide range level are redundant

(a) A minimum of either 1 wide range steam generator level plus 1 auxiliary feed-water flow channel per steam generator, or 2 wide range steam generator level channels per steam generator

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure - Narrow Range	M	R
2. Containment Pressure - Wide Range	M	R
3 ~~2.~~ Reactor Coolant Outlet Temperature - T_{Hot} (Wide Range)	M	R
4 ~~3.~~ Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	M	R
5 ~~4.~~ Pressurizer Pressure	M	R
6 ~~5.~~ Pressurizer Water Level	M	R
7 ~~6.~~ Steam Line Pressure	M	R
8 ~~7.~~ Steam Generator Water Level - Narrow Range	M	R
9 ~~8.~~ Steam Generator Water Level - Wide Range	M	R
10 ~~9.~~ Refueling Water Storage Tank Water Level	M	R
11 ~~10.~~ Auxiliary Feedwater Flow Rate	M	R
12 ~~11.~~ Reactor Coolant System Subcooling Margin Monitor	M	R
~~12. PORV Position Indicator~~	~~M~~	~~R~~
~~13. PORV Block Valve Position Indicator~~	~~M~~	~~R~~
13 ~~14.~~ Safety Valve Position Indicator	M	R
14 ~~15.~~ Spray System Temperature	M	R
15 ~~16.~~ Spray System Pressure	M	R
16 ~~17.~~ LPSI Header Temperature	M	R
17 ~~18.~~ Containment Temperature	M	R
18 ~~19.~~ Containment Water Level - Narrow Range	M	R
19 ~~20.~~ Containment Water Level - Wide Range	M	R
20 ~~21.~~ ~~Incore~~ Thermocouples Core Exit	M	R

NO PORV'S

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INSTRUMENTATION

FIRE DETECTION INSTRUMENTATION

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LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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~~5).

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.7.3 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.

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
1	Containment						
	Cable Tray Areas Elev 63'3"		6	1			
	Cable Tray Areas Elev 45'		4				
	Cable Tray Areas Elev 30'		4				
	Combustible Oil Area						
	Two steam generator rooms				32		32
	Charcoal Filter Area						
	Elev 45'				2		
2	Penetration						
	Elev 63'6"			12			
4	New Fuel Storage Area and Spent Fuel Pool Areas				(21)		
5	Control Building Elev 70'						
	Cable Riser Gallery Rm 423			2	24		
	Cable Riser Gallery Rm 449			3	24		
6	Control Building Elev 70'						
	Radiation Chemical Lab Rms 421, 420	1					
7	Radwaste Elev 63'6"						
	Chemical Storage Area Rm 503			1			
	Radwaste Control Panel Rm 513			1			
	Storage Area Rm 523			1			
	Hot Machine Shop	1					
8	Radwaste Elev 63'6"						
	Waste Decay Tank Rms 511A	None					
9	Fuel Handling Building Elev 45'						
	Emgy. A.C. Unit Rm 309-Train A			1	1		
	Emgy. A.C. Unit Rm 301-Train B			1	1		
10	Penetration						
	Elev 45'			6			

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

**Only zones with safety-related equipment are listed.

JUN 0 9 1981

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
1	Containment						
	Cable Tray Areas Elev 63'3"		6	1			
	Cable Tray Areas Elev 45'		4				
	Cable Tray Areas Elev 30'		4				
	Combustible Oil Area						
	Two steam generator rooms				32		32
	Charcoal Filter Area						
	Elev 45'				2		
2	Penetration						
	Elev 63'6"			12			
4	New Fuel Storage Area and						
	Spent Fuel Pool Areas	None			~~(21)~~	correction	
5	Control Building Elev 70'						
	Cable Riser Gallery Rm 423			2	24		
	Cable Riser Gallery Rm 449			3	24		
6	Control Building Elev 70'						
	Radiation Chemical Lab Rms 421, 420	1					
7	Radwaste Elev 63'6"						
	Chemical Storage Area Rm 503			1			
	Radwaste Control Panel Rm 513			1			
	Storage Area Rm 523			1			
	Hot Machine Shop	1					
8	Radwaste Elev 63'6"						
	Waste Decay Tank						
	Rms 511A	None					
9	Fuel Handling Building Elev 45'						
	Emgy. A.C. Unit Rm 309-Train A			1	1		
	Emgy. A.C. Unit Rm 301-Train B			1	1		
10	Penetration						
	Elev 45'			6			

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

**Only zones with safety-related equipment are listed.

JUL 0 7 1981
~~6/30/81~~
~~JUN 0 5 1981~~

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
11	S.E.B. Roof and Main Steam Relief Valves	None					
12	Control Building Elev 50'						
	Cable Riser Gallery Rm 305			3	42		
	Cable Riser Gallery Rm 315			3	40		
13A	Control Building Elev 30'						
	Emgy. HVAC Unit Rm 309A	1					
13B	Control Building Elev 50'						
	Emgy. HVAC Unit Rm 309B	1					
14	Radwaste Elev 24'						
	Boric Acid Makeup Tank Rm 204B	None					
	Boric Acid Makeup Tank Rm 204A	None					
15	Control Building Elev 50'						
	ESF Switchgear Rm 308A			2			
	ESF Switchgear Rm 308B			2			
16	Radwaste Elev 37' & 50'						
	Ion Exchangers	None					
17	Diesel Generator Building						
	Train A			3	4		
	Train B			3	4		
18	Diesel Fuel Oil Storage Tank Underground Vaults	None					
20	Condensate Storage Tank T-121	None					
21	Nuclear Storage Tank T-104	None					
22	Auxiliary Feedwater Pump Room			2			
23	Fuel Handling Bldg Elev 30'						
	Spent Fuel Pools Heat Exchange Room 209	None					
28	Penetration Elev. 30'			~~2~~	2		

correction

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
11	S.E.B. Roof and Main Steam Relief Valves	None					
12	Control Building Elev 50'						
	Cable Riser Gallery Rm 305			3	42		
	Cable Riser Gallery Rm 315			3	40		
13A	Control Building Elev 30'						
	Emgy. HVAC Unit Rm 309A	1					
13B	Control Building Elev 50'						
	Emgy. HVAC Unit Rm 309B	1					
14	Radwaste Elev 24'						
	Boric Acid Makeup Tank Rm 204B	None					
	Boric Acid Makeup Tank Rm 204A	None					
15	Control Building Elev 50'						
	ESF Switchgear Rm 308A			2			
	ESF Switchgear Rm 308B			2			
16	Radwaste Elev 37' & 50'						
	Ion Exchangers	None					
17	Diesel Generator Building						
	Train A			3	4		
	Train B			3	4		
18	Diesel Fuel Oil Storage Tank Underground Vaults	None					
20	Condensate Storage Tank T-121	None					
21	Nuclear Storage Tank T-104	None					
22	Auxiliary Feedwater Pump Room			2			
23	Fuel Handling Bldg Elev 30'						
	Spent Fuel Pools Heat Exchange Room 209	None					
28	Penetration Elev. 30'			2			

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning HEAT	Early Warning FLAME	Early Warning SMOKE	Actuation HEAT	Actuation FLAME	Actuation SMOKE
29	Control Building Elev 30'						
	Cable Riser Gallery Rm 236			3	51		
	Cable Riser Gallery Rm 224			3	52		
30	Electrical Tunnel Elev 30'6"			13	50		
31	Control Building Elev 30'			28			
32A	Control Building Elev 30'						
	Fan Room Rm 219 & Corridor Rm 221			1	2		
32B	Control Building Elev 30'						
	Fan Room Rm 233 & Corridor Rm 234			1			
34	Radwaste Elev 9' & 24'						
	Secondary Radwaste Tank Rms 126A,B & 127A,B	None					
35	Radwaste Elev 9' & 24'						
	Spent Resin Tank Rms 125A,B	None					
36	Fuel Handling Building Elev 17'6"						
	Spent Fuel Pool Pump Rm 107			2			
37	Radwaste Elev 24'						
	Letdown Heat Exchanger Rms 209A,B	None					
38	Radwaste Elev 24'						
	Letdown Control Valve Rms 218A,B	None					
39	Radwaste Elev 24'						
	Filter Crvd Tank Rm 216	None					
40	Radwaste Elev 9' & 24'						
	Primary Radwaste Tank Rms 211A,D	None					
41	Control Building Elev 9'						
	Cable Spreading Rm 111A			17	36		
	Cable Spreading Rm 111B			14	36		
42	Control Building Elev 9'						
	Cable Riser Gallery Rm 110			6	44		
	Cable Riser Gallery Rm 112			6	39		

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
43	Control Building Elev 9'						
	Emgy. Chiller Rm 115			2			
	Emgy. Chiller Rm 117			2			
44	Intake Structure						
	Pump Rm T2-106			4			
	Pump Rm T3-106			4			
45	Containment Elev 9' & 15'						
	Piping Penetration Area 15'		4				
48	Safety Equipment Building 9'						
	CCW HX and Piping Rm 022-025	None					
50	Radwaste Elev 9'						
	Charging Pump Rms 106A-F			6			
51	Radwaste Elev 9'						
	Boric Acid Makeup Tank Rms 105A-D	None					
53	Electrical Tunnel Elev 9'6"			21	54		
54	Safety Eqpmnt Bldg Elev 15'6" & 8'						
	Shutdown HX Rms 003, 004, 016, 018	None					
55	Safety Eqpmt Bldg Elev 8'						
	Chemical Storage Tank Rm 019			1			
56	Safety Eqpmt Bldg Elev 8'						
	Component Cooling Water Surge Tank Rms 020, 021	None					
57	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 005			1			
58	Radwaste Elev 37'						
	Reactor Trip System Rms 308A-D, 309A-C			9			
59	Safety Eqpmt Bldg Elev 15'6"						
	Pump Rm 001			1			

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
60	Safety Eqpmt Bldg Elev 15'6" Pump Rm 015			1			
61	Safety Eqpmt Bldg Elev 15'6" Component Cooling Water Pump Rms 006, 007, 008			3			
62	Radwaste Elev 50' Volume Control Valve Rooms	None					
63	Control Building Elev 50' Corridor			12			
64	Control Building Elev 50' Vital Power Distribution Rms 310A-H			8			
65	Control Building Elev 50' Battery Rms 306B-J			8			
66	Control Building Elev 50' Evacuation Rm 311			1			
67	Radwaste Elev 63'6"						
	Cable Riser Gallery Rm 506A			2	4		
	Cable Riser Gallery Rm 506B			2	4		
68	Penetration 9' - 63'6" Cable Riser Shaft			1	21		
69	Safety Eqpmt Bldg Elev 5'3" Salt Water Cooling Piping Rm 010	None					
70	Radwaste Elev 24' Duct Shaft Rms 222A,B	None					
72	Control Building Elev 70' Corridor 442	None					
75	Refueling Water Storage Tank T-005	None					
76	Refueling Water Storage Tank T-006	None					

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TABLE 3.3-11

FIRE DETECTION INSTRUMENTS
MINIMUM INSTRUMENTS OPERABLE*

Zone**	Instrument Location	Early Warning			Actuation		
		HEAT	FLAME	SMOKE	HEAT	FLAME	SMOKE
78	Control Building Elev 9' Corridor Rm 105			 4			
79	Control Building Elev 50' ESF Switchgear Rm 302A ESF Switchgear Rm 302B			 2 2			
80	Radwaste Elev 37' & 50' Duct Shaft Rms	 None					
81	Radwaste Elev 63'6" Duct Shaft Rms 527A,B	 None					
83	Salt Water Cooling Tunnel			6			
84	Safety Eqpmt Bldg Elev 8' HVAC Rm 017			 3			

INSTRUMENTATION DRAFT

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12.

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	(1)	20 ~~28~~
b. Neutralization Sump ~~Steam Generator Blowdown~~ Effluent Line	(1)	21 ~~29~~
c. Turbine Building ~~(Floor Drains)~~ Sumps Effluent Line	(1)	22 ~~30~~
~~2. GROSS RADIOACTIVITY MONITORS NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE~~		
~~a. Service Water System Effluent Line~~	~~(1)~~	~~30~~
~~b. Component Cooling Water System Effluent Line~~	~~(1)~~	~~30~~
~~3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR~~		
~~a. Steam Generator Blowdown Effluent Line~~	~~(1)~~	~~29~~
~~b. Turbine Building Sumps Effluent Line~~	~~(1)~~	~~30~~
2. ~~4.~~ FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	(1)	23 ~~31~~
~~b. Discharge Canal~~	~~(1)~~	~~31~~
b ~~c.~~ Neutralization Sump ~~Steam Generator Blowdown~~ Effluent Lines	(1)	23 ~~31~~

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TABLE 3.3-12 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
5. RADIOACTIVITY RECORDERS (*)		
a. Liquid Radwaste Effluent Line	(1)	33
b. Steam Generator Blowdown Effluent Line	(1)	34
6. TANK LEVEL INDICATING DEVICES (for tanks outside plant buildings)		
a. ____________	(1)	32
b. ____________	(1)	32
c. ____________	(1)	32
d. ____________	(1)	32

(*Required only if alarm/trip set point is based on recorder-controller)

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TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION ~~28~~ 20 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases may continue for up to 14 days provided that prior to initiating a release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION ~~29~~ 21 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/gram:

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

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

ACTION ~~30~~ 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided that, at least once per 8 hours, grab samples are collected and analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 10^{-7} microcuries/ml.

ACTION ~~31~~ 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours during actual releases. Pump curves may be used to estimate flow.

~~ACTION 32 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, liquid additions to this tank may continue for up to 30 days provided the tank liquid level is estimated during all liquid additions to the tank.~~ N/A

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TABLE 3.3-12 (Continued)

TABLE NOTATION

ACTION 33 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 14 days provided the gross radioactivity level is determined at least once per 4 hours during actual releases.

ACTION 34 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided the gross radioactivity level is determined at least once per 4 hours during actual release.

N/A

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

renumber to reflect N/A & existing note 2

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluents Line	D	P	R(~~3~~ 2)	Q(1)
b Neutralization Sump ~~b. Steam Generator Blowdown~~ Effluent Line	D	M	R(~~3~~ 2)	Q(1)
c. Turbine Building ~~(Floor Drains)~~ Sumps Effluent Line	D	M	R(~~3~~ 2)	Q(1)
~~2. GROSS BETA OR GAMMMA RADIOACTIVITY MONITORS PROVIDING ALARM BUT NOT PROVIDING AUTOMATIC TERMINATION OF RELEASE~~				
~~a. Service Water System Effluent Line~~	~~D~~	~~M~~	~~R(3)~~	~~Q(2)~~
~~b. Component Cooling Water System Effluent Line~~	~~D~~	~~M~~	~~R(3)~~	~~Q(2)~~
~~3. CONTINUOUS COMPOSITE SAMPLERS AND SAMPLER FLOW MONITOR~~				
~~a. Steam Generator Blowdown Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~b. Turbine Building Sumps Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~

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TABLE 4.3-8 (Continued)

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

renumbered notes

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
4. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D(~~4~~) 3	N.A.	R	Q
b. ~~Steam Generator Blowdown~~ Neutralization sump Effluent Line	D(~~4~~) 3	N.A.	R	Q
~~c. Discharge Canal~~	~~D(4)~~	~~N.A.~~	~~R~~	~~Q~~
~~5. RADIOACTIVITY RECORDERS~~				
~~a. Steam Generator Blowdown Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~b. Liquid Radwaste Effluent Line~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~
~~6. TANK LEVEL INDICATING DEVICES (for tanks outside the building)~~				
~~a. ________~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~
~~b. ________~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~
~~c. ________~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~
~~d. ________~~	~~D*~~	~~N.A.~~	~~R~~	~~Q~~

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TABLE 4.3-8 (Continued)

TABLE NOTATION

~~* During liquid additions to the tank.~~ N/A

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

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

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

~~1. Instrument indicates measured levels above the alarm setpoint.~~

~~2. Circuit failure.~~

~~3. Instrument indicates a downscale failure.~~

~~4. Instrument controls not set in operate mode.~~

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

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

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RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-13

ACTION:

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13.

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

SURVEILLANCE REQUIREMENTS

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

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TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

Notes renumbered

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS HOLDUP SYSTEM			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (High Range only)	~~(1)~~ 1	*	~~35~~ 24
~~b. Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
~~d.~~ b Effluent System Flow Rate Measuring Device	~~(1)~~ 1	*	~~36~~ 25
~~e. Sampler Flow Rate Measuring Device~~	~~(1)~~	~~*~~	~~36~~
~~2A. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM (for systems designed to withstand the effects of a hydrogen explosion)~~			
~~a. Hydrogen Monitor~~	~~(1)~~	~~**~~	~~39~~
~~b. Hydrogen or Oxygen Monitor~~	~~(1)~~	~~**~~	~~39~~
~~2B.~~ 2 WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM ~~(for systems not designed to withstand the effects of a hydrogen explosion)~~			
a. Hydrogen Monitor	~~(2)~~ 2	**	~~39~~ ~~40~~ 28
b. ~~Hydrogen or~~ Oxygen Monitor	~~(2)~~ 2	**	~~39~~ ~~40~~ 28

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TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
3. CONDENSER AIR EJECTOR ~~EVACUATION SYSTEM~~			
a. Noble Gas Activity Monitor	~~(1)~~ 1	*	~~37~~ 26
~~b. Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
*** ~~d.~~ Flow Rate Monitor	~~(1)~~ 1	*	~~36~~ 25
~~e. Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
4. Plant Vent Stack Airborne Monitor ~~VENT HEADER SYSTEM~~			
a. Noble Gas Activity Monitor	~~(1)~~ 1	*	~~37~~ 26
b. Iodine Sampler	~~(1)~~ 1	*	29 ~~41~~ 40
c. Particulate Sampler	~~(1)~~ 1	*	29 ~~41~~ 40
*** d. Flow Rate Monitor	~~(1)~~ 1	*	25 ~~36~~
~~e. Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
5. CONTAINMENT PURGE EXHAUST ~~SYSTEM~~			
*** a. Noble Gas Activity Monitor - Providing Alarm ~~and Automatic Termination of Release~~	(1)	*	~~38~~ 27
~~b. Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~

~~** NUREG 0737 item to be operational 7/1/82~~

TABLE 3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. CONTAINMENT PURGE EXHAUST ~~SYSTEM~~ (Continued)			
** b. ~~e.~~ Flow Rate Monitor	(1)	*	25 ~~36~~
~~f. Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~6. AUXILIARY BUILDING VENTILATION SYSTEM~~			
~~a. Noble Gas Activity Monitor~~	~~(1)~~	~~*~~	~~37~~
~~b. Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
~~d. Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~e. Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~7. FUEL STORAGE AREA VENTILATION SYSTEM~~			
~~a. Noble Gas Activity Monitor~~	~~(1)~~	~~*~~	~~37~~
~~b. Iodine Sampler~~	~~(1)~~	~~*~~	~~41~~
~~c. Particulate Sampler~~	~~(1)~~	~~*~~	~~41~~
~~d. Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~
~~e. Sampler Flow Rate Monitor~~	~~(1)~~	~~*~~	~~36~~

~~** NUREG 0737 ITEM TO BE OPERATIONAL 1/1/82~~

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TABLE 3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
8. RADWASTE AREA VENTILATION SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
e. Sampler Flow Rate Monitor	(1)	*	36
9. STEAM GENERATOR BLOWDOWN VENT SYSTEM			
a. Noble Gas Activity Monitor	(1)	*	37
b. Iodine Sampler	(1)	*	41
c. Particulate Sampler	(1)	*	41
d. Flow Rate Monitor	(1)	*	36
e. Sampler Flow Rate Monitor	(1)	*	36

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TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

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

*** NUREG-0737 item to be operational by January 1, 1982

ACTION ~~35~~ 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, the contents of the tank(s) may be released to the environment for up to 14 days provided that prior to initiating the release:

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

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

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION ~~36~~ 25 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided the flow rate is estimated at least once per 4 hours.

ACTION ~~37~~ 26 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are taken at least once per 8 hours and these samples are analyzed for gross activity within 24 hours.

ACTION ~~38~~ 27 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, immediately suspend PURGING of radioactive effluents via this pathway.

~~ACTION 39 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, operation of this waste gas holdup system may continue for up to 30 days provided grab samples are collected at least once per 4 hours and analyzed within the following 4 hours.~~

ACTION ~~40~~ ~~29~~ 28 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 14 days. With (two) channels inoperable, be in at least HOT STANDBY within 6 hours.

ACTION ~~41~~ ~~40~~ 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via the affected pathway may continue for up to 30 days provided samples are continuously collected with auxiliary sampling equipment as required in Table 4.11-2.

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. WASTE GAS HOLDUP SYSTEM					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release (High Range only)	P	P	R(3)	Q(1)	*
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
b ~~d.~~ Flow Rate Monitor	P	N.A.	R	Q ~~Q~~	* ~~*~~
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
2. WASTE GAS HOLDUP SYSTEM EXPLOSIVE GAS MONITORING SYSTEM					
a. Hydrogen Monitor	D	N.A.	Q(4)	M	**
b. Hydrogen Monitor (alternate)	D	N.A.	Q(4)	M	**
c. Oxygen Monitor	D	N.A.	Q(5)	M	**
d. Oxygen Monitor (alternate)	D	N.A.	Q(5)	M	**

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
3. CONDENSER AIR EJECTOR ~~EVACUATION SYSTEM~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
(*** b ~~d.~~ Flow Rate Monitor	D	N.A.	R	Q	*)
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
4. PLANT VENT STACK ~~VENT HEADER SYSTEM~~					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
(*** d. Flow Rate Monitor	D	N.A.	R	Q	*)
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~

~~** NUREG 0737 item to be operational 1/1/81~~

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
5. CONTAINMENT PURGE EXHAUST ~~SYSTEM~~					
*** a. Noble Gas Activity Monitor - Providing Alarm ~~and Automatic Termination of Release~~	D	P	R(3)	Q(1)	*
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
*** d. Flow Rate Monitor	D	N.A.	R	Q	*
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~6. AUXILIARY BUILDING VENTILATION SYSTEM~~					
~~a. Noble Gas Actvity Monitor~~	~~D~~	~~M~~	~~R(3)~~	~~Q(2)~~	~~*~~
~~b. Iodine Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~c. Particulate Sampler~~	~~W~~	~~N.A.~~	~~N.A.~~	~~N.A.~~	~~*~~
~~d. Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~
~~e. Sampler Flow Rate Monitor~~	~~D~~	~~N.A.~~	~~R~~	~~Q~~	~~*~~

~~** NUREG 0737 item. To be operational by 1/1/82~~

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TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
7. FUEL STORAGE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*
8. RADWASTE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
9. STEAM GENERATOR BLOWDOWN VENT					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Flow Rate Monitor	D	N.A.	R	Q	*
e. Sampler Flow Rate Monitor	D	N.A.	R	Q	*

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TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times.

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

*** NUREG 0737 item to be operational by January 1, 1982

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

1. Instrument indicates measured levels above the alarm/trip setpoint.
2. Circuit failure.
3. Instrument indicates a downscale failure.
4. Instrument controls not set in operate mode.

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

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

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

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

1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

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

1. One volume percent oxygen, balance nitrogen, and
2. Four volume percent oxygen, balance nitrogen.

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INSTRUMENTATION

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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 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~~ Six low pressure turbine reheat stop valves.
4. ~~Four~~ Six 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.

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.

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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 Both Reactor Coolant loops and associated steam generator and both Reactor Coolant pumps in each loop shall be in operation.

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.

*See Special Test Exception 3.10.3.

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LIMITING CONDITION FOR OPERATION

3.4.1.2 a. The Reactor Coolant loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump.

2. Reactor Coolant Loop 2 and its associated steam generator 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 Reactor Coolant loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be ≥ (23%) 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.

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REACTOR COOLANT SYSTEM

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 a. At least two of the Reactor Coolant and/or shutdown cooling loops listed below shall be OPERABLE:

1. Reactor Coolant Loop 1 and its associated steam generator and at least one associated Reactor Coolant pump,**

2. Reactor Coolant Loop 2 and its associated steam generator and at least one associated Reactor Coolant pump,**

3. Shutdown Cooling Train A,

4. Shutdown Cooling Train B.

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

APPLICABILITY: MODE 4#

ACTION:

a. With less than the above required Reactor Coolant and/or shutdown cooling loops OPERABLE, immediately initiate corective action to return the required loops to OPERABLE status as soon as possible, be in COLD SHUTDOWN within 20 hours.

b. With no Reactor Coolant or shutdown cooling 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.

#With the Reactor Coolant System temperature less than or equal to 280°F, the shutdown cooling system shall be in operation with all suction line valves open.

All Reactor Coolant pumps and shutdown cooling pumps may be de-energized up for 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 if the associated steam generator to Reactor Coolant System ΔT is greater than 100°F.

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REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

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 ≥ ____________ at least once per 12 hours.

4.4.1.3.4 At least one Reactor Coolant or shutdown cooling loop shall be verified to be in operation and circulating Reactor Coolant at least once per 12 hours.

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

omitted previously

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LIMITING CONDITION FOR OPERATION

3.4.1.4 Two# shutdown cooling ~~loops~~ trains shall be OPERABLE* and at least one shutdown cooling ~~loop~~ train shall be in operation.**

APPLICABILITY: MODE 5##.

ACTION:

a. With less than the above required shutdown cooling trains/Reactor Coolant loops OPERABLE, immediately initiate corrective action to return the required shutdown cooling trains/Reactor Coolant loops to OPERABLE status as soon as possible.

b. With no shutdown cooling ~~loop~~ train 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 shutdown cooling ~~loop~~ train to operation.

SURVEILLANCE REQUIREMENTS

4.4.1.4 The ~~residual heat removal loop~~ shutdown cooling train shall be determined to be in operation and circulating reactor coolant at least once per 12 hours.

#One shutdown cooling ~~loop~~ train may be inoperable for up to 2 hours for surveillance testing provided the other shutdown cooling ~~loop~~ train is OPERABLE and in operation. Two filled Reactor Coolant loops, each with a steam generator secondary side water level of greater than or equal to (23%), may be substituted for one shutdown cooling ~~loop~~ train.

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##A Reactor Coolant pump shall not be started if the associated steam generator to Reactor Coolant System ΔT is greater than 100°F.

*The normal or emergency power source may be inoperable.

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

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REACTOR COOLANT SYSTEM

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3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA ± 1%.* whenever the shutdown cooling system is not in operation.

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

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

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

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LIMITING CONDITION FOR OPERATION

3.4.2.2 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

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

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

[redacted]

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with ~~at least ___ Kw of pressurizer heaters~~ a water volume of less than or equal to ___ cubic feet, and two groups of pressurizer heaters each having a capacity of at least ___ kw.

(later) CE to provide

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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3/4.4.4 RELIEF VALVES

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.4.1 In addition to the requirements of Specification 4.0.5, each PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION and by operating the valve through one complete cycle of full travel.

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

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

no PORV's in SO 2/3

~~[illegible]~~

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REACTOR COOLANT SYSTEM

3/4.4.~~5~~4 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.~~5~~4 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.~~5~~4.0 Each steam generator shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

4.4.~~5~~4.1 Steam Generator Sample Selection and Inspection - Each steam generator shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of steam generators specified in Table 4.4-1.

4.4.~~5~~4.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.~~5~~4.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.~~5~~4.4. 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.~~5~~4.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

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

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

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

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

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

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

4.4.~~5~~4.3 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~~ Syncronization. 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 both 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~~ not less than 12 nor more than 24 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.4.5.3.a; the interval may then be extended to a maximum of once per 40 months.

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

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

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4.4.~~5~~ 4.4 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.

not used ~~4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.~~

4 ~~5.~~ Defect means an imperfection of such severity that it exceeds the plugging limit ~~. A tube containing a defect is defective.~~ shown in figure 4.4-1.

see #5 above ~~6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service and is equal to (40)%* of the nominal tube wall thickness, as determined in Figure 4.4-1~~

not used ~~7. Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.4.5.3.c, above.~~

5 ~~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.

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

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

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SURVEILLANCE REQUIREMENTS (Continued)

of the tubing. This inspection Was ~~shall be~~ performed Prior to ~~after the~~ field hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections. Q 121.27

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

4.4.~~5~~4.5 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~~ 14 days. Q 121.27

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.

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MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

No. of Steam Generators per Unit .... Two
First Inservice Inspection .... One
Second & Subsequent Inservice Inspections .... One*

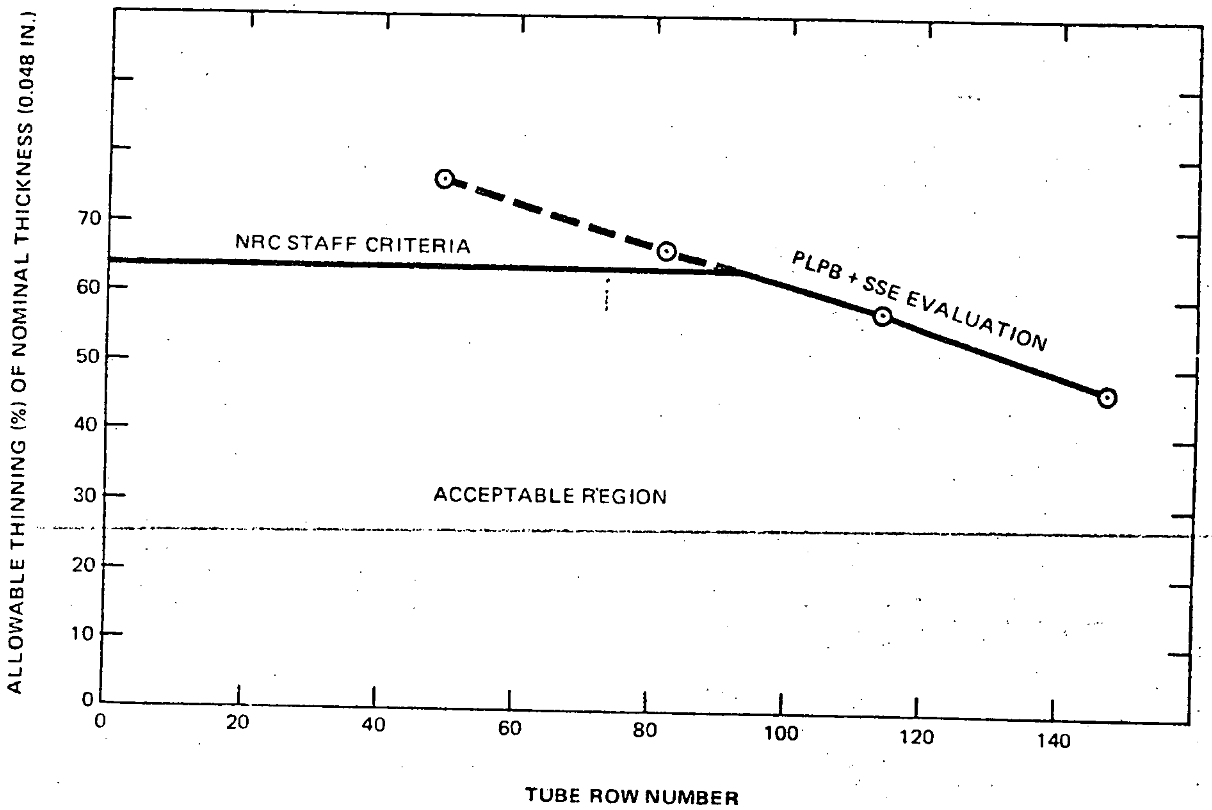
*The inservice inspection may be limited to one steam generator on a rotating schedule encompassing 3N% of the tubes (where N is the number of steam generators in the plant) if the results of the first or previous inspections indicate that all steam generators are performing in a like manner. Note that 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.

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



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

TUBE WALL THINNING
ACCEPTANCE CRITERIA

Figure 4.4-1

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REACTOR COOLANT SYSTEM

3/4.4.~~6~~5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.~~6~~5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump inlet flow monitoring system, and

c. ~~The~~ A containment atmosphere gaseous radioactivity monitoring system.

Also covered in Spec 3/4.3.3.1

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.~~6~~5.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

consistent FSAR 5.2.5

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REACTOR COOLANT SYSTEM

3/4.4.~~6~~5 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.~~6~~5.1 The following Reactor Coolant System leakage detection systems shall be OPERABLE:

a. A containment atmosphere particulate radioactivity monitoring system,

b. The containment sump inlet flow monitoring system, and

c. The containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous 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.~~6~~5.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

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REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.~~6~~5.2 Reactor Coolant System leakage shall be limited to:

no loop isolation valves

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

Not Applicable (no seal injection)

~~e. ___ gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.~~

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

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.~~6~~5.2 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

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

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

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REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.~~6~~5.2 Reactor Coolant System leakage shall be limited to:

correct # for 2 S/G per unit

720

no loop isolation valves

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

Not Applicable (no seal injection)

~~e. ___ gpm CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2235 ± 20 psig.~~

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

APPLICABILITY: MODES 1, 2, 3 and 4

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.4.~~6~~5.2.1 Reactor Coolant System leakages shall be demonstrated to be within each of the above limits by:

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

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

consistency w/ 3/4.4.5.1

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SURVEILLANCE REQUIREMENTS (Continued)

Not applicable (no seal injection)

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

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

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

See insert below

~~4.4.6.2 Each Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1 shall be demonstrated OPERABLE pursuant to Specification 4.0.5 except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit.~~

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

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

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

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

N/A Spec 3.0.3/3.0.4 ??

4.4.5 ~~6~~.2.2 Each Reactor Coolant System Pressure Isolation valve specified in Tables 3.4-1a and 3.4-1b 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. After returning the valve to service following maintenance, repair or replacement work on the valve: within 24 hours if in modes 1,2, or 3, otherwise within 24 hours after entry into mode 3.

c. For valves specified in Table 3.4-1b ~~a~~, following valve actuation due to automatic or manual action or flow through the valve: within 24 hours if in modes 1,2, or 3, otherwise within 24 hours after entry into mode 3.

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TABLE 3.4-1a
REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES
(VALVES ISOLATING SYSTEMS RATED ABOVE 50% OF RCS DESIGN PRESSURE)

3-018-A-551	HPSI Check
3-019-A-551	HPSI Check
3-020-A-551	HPSI check
3-021-A-551	HPSI Check
3-152-A-551	Hot leg injection to loop #1
3-156-A-551	Hot leg injection to loop #2
3-157-A-551	Hot leg injection check
3-158-A-551	Hot leg injection check

TABLE ~~4.5~~ 3.4-1b
(VALVES ISOLATING SYSTEMS RATED BELOW 50% OF RCS DESIGN PRESSURE)

8-072-A-552	LPSI Check
8-073-A-552	LPSI Check
8-074-A-552	~~LPST~~ LPSI Check
8-075-A-552	~~LPST~~ LPSI Check
12-027-A-551*	Cold leg injection to loop #1A
12-029-A-551*	Cold leg injection to loop #1B
12-031-A-551*	Cold leg injection to loop #2A
12-033-A-551*	Cold leg injection to loop #2B
12-040-A-551	SIT Check
12-041-A-551	SIT Check
12-~~041~~ 042-A-551	SIT Check
12-043-A-551	SIT Check
2 HV-9337	SDC Suction Isolation
2 HV-9339	SDC Suction Isolation
2HV-9377	SDC Suction Isolation
2 ~~M~~ H V-9378	SDC Suction Isolation

*Redundant to LPSI and SIT checks

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3/4.4.~~7~~6 CHEMISTRY

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LIMITING CONDITION FOR OPERATION

3.4.~~7~~6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

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

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

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 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.~~7~~6 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.

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3/4.4.~~7~~6 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.~~7~~6 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-2.

APPLICABILITY: At all times.

ACTION:

MODES 1, 2, 3 and 4:

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

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

At All Other Times:

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to less than or equal to 500 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.~~7~~6 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.

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TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

PARAMETER	STEADY STATE LIMIT	TRANSIENT LIMIT
DISSOLVED OXYGEN*	≤ 0.10 ppm	≤ 1.00 ppm
CHLORIDE	≤ 0.15 ppm**	≤ 1.50 ppm
FLUORIDE	≤ 0.15 ppm	≤ 1.50 ppm

*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.

to permit initial fill of refueling canal each refueling

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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

[redacted]

REACTOR COOLANT SYSTEM

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3/4.4.~~8~~ 7 SPECIFIC ACTIVITY

note: 6.5 µCi/gm is in accordance with Response to NRC Question 312.35

LIMITING CONDITION FOR OPERATION

3.4.~~8~~ 7 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 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 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 these circumstances does not exceed 800 hours in any consecutive 12 month period. With the total cumulative operating time at a primary coolant specific activity greater than 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 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 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ 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 6.5 microcurie/gram DOSE EQUIVALENT I-131.

SURVEILLANCE REQUIREMENTS

4.4.~~8~~7 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	At least once per 72 hours	1, 2, 3, 4
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	1 per 14 days	1
3. Radiochemical for $\bar{E}$ Determination	1 per 6 months*	1
4. Isotopic Analysis for Iodine Including I-131, I-133, and I-135	a) Once per 4 hours, whenever the specific activity exceeds 6.5 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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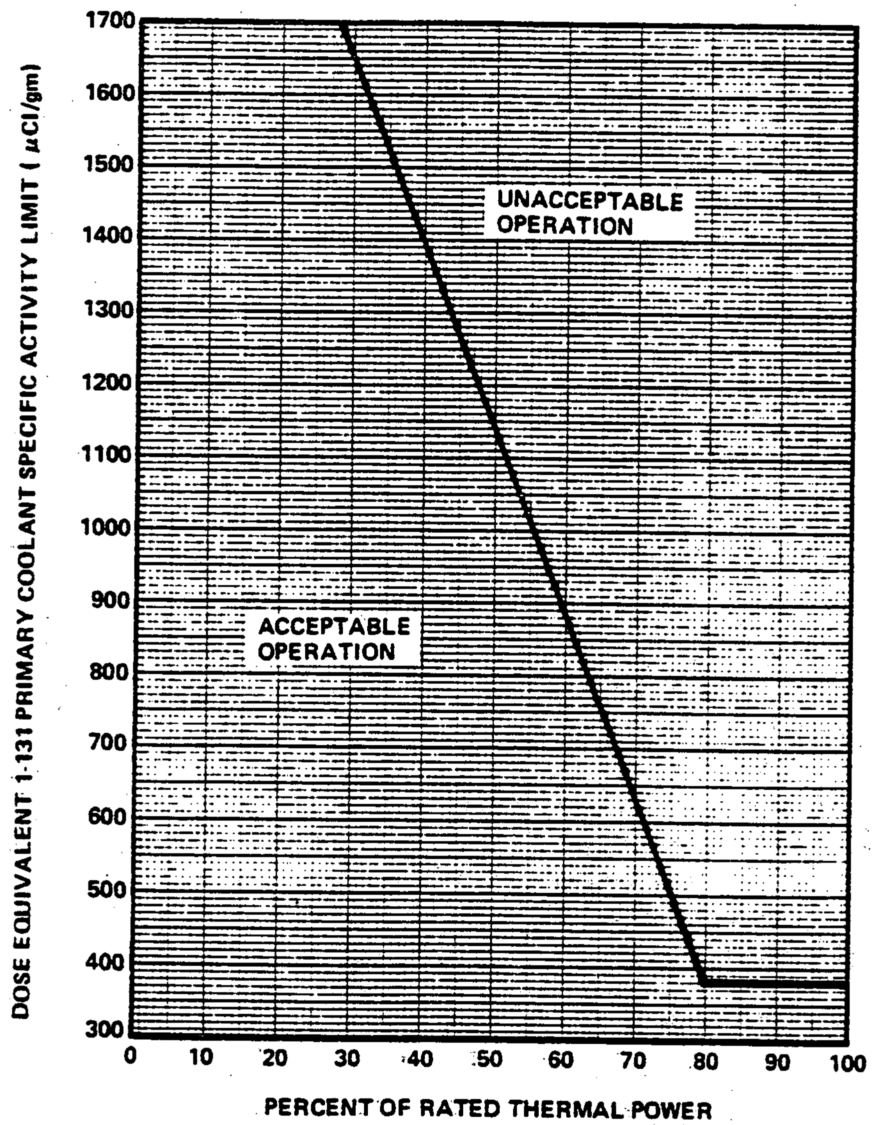

Figure 3.4-1

DOSE EQUIVALENT 1-131 PRIMARY COOLANT SPECIFIC ACTIVITY LIMIT VERSUS PERCENT OF RATED THERMAL POWER WITH THE PRIMARY COOLANT SPECIFIC ACTIVITY 6.5 μ Ci/gram DOSE EQUIVALENT

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REACTOR COOLANT SYSTEM

3/4.4.~~9~~8 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.~~9~~8.1 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 60°F in any one hour period.

b. A maximum cooldown of 100°F in any one hour period 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.~~9~~8.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.~~9~~8.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME
1	83°	1.15	Standby
2	97°	1.15	4 EFPY
3	104°	1.15	17 EFPY
4	284°	1.15	30 EFPY
5	263°	1.15	Standby
6	277°	1.15	Standby

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Figure 3.4-2

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LIMITING CONDITION FOR OPERATION

3.4.8.2 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 of 1000 spray cycles of spray water temperature differential of ~~greater than~~ 200°F. per section 5.7

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

4.4.8.2 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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OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. The Shutdown Cooling System (SDCS) Relief Valve (PSV9349) with a lift setting of less than or equal to ~~(450)~~ (later) psig, or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to (1.3) square inches.

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

ACTION:

a. With the SDCS Relief Valve inoperable, depressurize and vent the RCS through a greater than or equal to (1.3) square inch vent(s) within the next 8 hours.

b. In the event either the SDCS Relief Valve or an RCS vent is 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 SDCS Relief Valve or RCS vent on the transient and any corrective action necessary to prevent recurrence.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.8.3.1 The SDCS Relief Valve shall be demonstrated OPERABLE by:

~~a. Verification of the SDCS Relief Valve setpoint at least once per 18 months.~~

No, see Q&R 212.141 and next page

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SURVEILLANCE REQUIREMENTS (Continued)

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

b. ~~c.~~ Testing pursuant to Specification 4.0.5. Inservice test intervals shall not exceed 30 months. per Q&R 212.141

4.4.~~6~~8.3.2 The RCS vent shall be verified to be open at least once per 12 hours* when the vent is being used for overpressure protection.

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* 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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3.4.~~10~~9 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.~~10~~9 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: ALL MODES

ACTION:

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

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

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component 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.~~10~~9 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.

REACTOR COOLANT SYSTEM

3.4.~~10~~9 STRUCTURAL INTEGRITY

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LIMITING CONDITION FOR OPERATION

3.4.~~10~~9 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.

APPLICABILITY: ALL MODES

ACTION:

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

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

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component 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.~~10~~9 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

3/4.5.1 SAFETY INJECTION TANKS

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LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open, and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1720 and 2300 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and ~~624~~ 625 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:

No! Use indications so have a record !!

1. Verifying ~~(by the absence of alarms)~~ the contained borated water volume and nitrogen cover-pressure in the tanks is within the above limits, 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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3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 SAFETY INJECTION TANKS

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LIMITING CONDITION FOR OPERATION

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

a. The isolation valve open, and power to the valve removed,

b. A contained borated water volume of between 1680 and 1807 cubic feet,

c. Between 1720 and 2300 ppm of boron, and

d. A nitrogen cover-pressure of between 600 and ~~624~~ 625 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 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.

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. When an actual or simulated RCS pressure signal exceeds 505 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 check ~~FUNCTIONAL TEST.~~

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

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EMERGENCY CORE COOLING SYSTEMS

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

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE 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.

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SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

Valve Number	Valve Function	Valve Position
~~a. 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

See attached

b. At least once per 31 days by:

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

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

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

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

2. Of the areas affected within containment at the completion of 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 to prevent opening of the Shutdown Cooling System isolation valves when RCS pressure is greater than or equal to 376 psia.

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Insert to 4.5.2.a:

a.	HV9353	SDC Warmup	CLOSED
b.	HV9359	SDC Warmup	CLOSED
c.	HV8150	SDC(HX) Isolation	CLOSED
d.	HV8151	SDC(HX) Isolation	CLOSED
e.	HV8152	SDC(HX) Isolation	CLOSED
f.	HV8153	SDC(HX) Isolation	CLOSED
g.	FV0306	SDC Bypass Flow Control	OPEN
h.	One of the following:		
1)	HV9316	SDC(HX) Flow Control	OPEN
or 2)	14-079	SDC(HX) Flow Control (Manual)	OPEN

to reflect interim changes to SDC

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EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on 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 Recirculation Actuation Test Signal, the containment sump isolation valves open and the recirculation ~~valves~~ valve to the refueling water tank ~~closed~~ closes.

2 series valve in each line

f. By verifying that each of the following pumps develops the indicated discharge pressure 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. ×

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[redacted]

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SURVEILLANCE REQUIREMENTS (Continued)

9x. 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 ____ gpm~~

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

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

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

(later)

CE to provide

LPSI System - Single Pump

~~a. Injection Leg 1, greater than or equal to ____ gpm~~ x

~~b. Injection Leg 2, greater than or equal to ____ gpm~~ x

~~c. Injection Leg 3, greater than or equal to ____ gpm~~ x

~~d. Injection Leg 4, greater than or equal to ____ gpm~~ x

(later)

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3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 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 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.

*With pressurizer pressure less than 1700 psia.

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3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water storage tank shall be OPERABLE with:

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

b. Between 1720 and 2300 ppm of boron, and

c. A minimum solution temperature of ~~35~~ 40°F. C-E

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The RWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is less than ~~35~~ 40°F. C-E

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

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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~~3.

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

c. After each closing of each penetration subject to Type B testing, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a (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.

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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, 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 (55.7 psig) or at P_t (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.

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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:

   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.

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

   1. Air locks, and

   2. 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. 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 (61.3 psig) and the seal system capacity is adequate to maintain system pressure for at least 30 days.

SURVEILLANCE REQUIREMENTS (Continued)

g. 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. The provisions of Specification 4.0.2 are not applicable.

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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, 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.

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SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. 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 {55.7 psig} for at least 15 minutes, X

b. Prior to establishing CONTAINMENT INTEGRITY if opened when CONTAINMENT INTEGRITY was not required, and at least once per 6 months# by conducting an overall air lock leakage test at P_a {55.7 psig} and by verifying that the overall air lock leakage rate is within its limit, and X

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

#The provisions of Specification 4.0.2 are not applicable.

INTERNAL PRESSURE

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LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between +1.5 and -0.3 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

AIR TEMPERATURE

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LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.5 The primary containment average air temperature shall be the arithmetical average of the temperatures at 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"

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CONTAINMENT SYSTEMS

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

4.6.1.6.1 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. 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 1393 (minimum) and 1644 (maximum) kips at the first year inspection and that a representative sample* of at least 4%, but no less than ~~6~~ 5, of the hoop tendons each have a lift off force of between 1281 (minimum) and 1626 (maximum) kips 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: 22.5 log t (U-tendon), 21.8 log t (hoop tendons) and 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) 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

per surveillance table

*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~~.
in accordance with Table 4.6-1.

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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~~ 3 and a representative sample of at least 2%, but no less than 3 of the hoop tendons.

per surveillance table

b. Performing a Detensioning and Material Test and Inspection by 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 270 ksi (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)

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

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

4.6.1.6.4 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.

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TABLE 4.6-1

TENDON SURVEILLANCE

Years After Initial Structural Integrity Test	TENDON NUMBERS									
	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		
Detensioning and Material Tests	20	88-154			42	19-133	97	66-176		

Years After Initial Structural Integrity Test	TENDON NUMBERS									
	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 VENTILATION SYSTEM

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LIMITING CONDITION FOR OPERATION

3.6.1.7 The containment large volume purge supply and exhaust isolation valves shall be closed with power to the valves removed.

to meet 0737 commitment to lock closed

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

ACTION:

With one containment large volume 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.

to clarify that requirement applies only to the large volume purge and not minipurge

SUREVILLANCE REQUIREMENTS

4.6.1.7 The containment large volume purge supply and exhaust isolation valves ~~shall be determined closed at least once per 31 days.~~

a. Shall be determined closed at least once per 24 hours.

NUREG 0737 II.E.4.2

b. Shall be verified to have power to the valves removed at least once per 31 days.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two independent containment spray systems shall be OPERABLE with each spray system capable of taking suction from the RWST on a Containment Spray Actuation Signal and automatically transferring suction to the containment sump on a 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, and 3 ~~and 4.~~

SDCHX not available for spray in Mode 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 RWST on a Containment Spray Actuation (CSAS) test signal.

b. By testing 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 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.

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THIS [illegible] RECEIPT OF INFO[illegible] [illegible]NDMENT

CONTAINMENT SYSTEMS

IODINE REMOVAL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The iodine removal system shall be OPERABLE with:

we to provide

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 chemical addition pumps each capable of adding NaOH solution from the chemical addition tank to a containment spray system pump flow.

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

ACTION:

With the 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 iodine removal system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the NaOH solution temperature.

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

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See spec 3.10.3 for case of 1 inoperable cooling fan in Mode 4 w/ both SDCHX lined up for SDC operation

CONTAINMENT COOLING SYSTEM

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.

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.

 DRAFT

See Δ# B-338 to FSAR

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

Manual valves are covered in 3/4.6.1.1

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 containment isolation valve(s) specified in Table 3.6-1 Sections A and B 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 (as applicable).

4.6.3.2 Each isolation valve specified in Table 3.6-1 Sections A and B 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 in Table 3.6-1 Section A actuates to its isolation position.

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in Table 3.6-1 Section B

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that on a Containment Radiation-High test signal, all containment purge valves actuate to their isolation position. ×

Section A

4.6.3.3 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. ×

4.6.3.4 Each containment 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.

[redacted]

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in Table 3.6-1 Section B

SURVEILLANCE REQUIREMENTS (Continued)

b. Verifying that on a Containment Radiation-High test signal, all containment purge valves actuate to their isolation position.

Section A

4.6.3.3 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.

4.6.3.4 Each containment 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 $0.60 L_a$.

18 HV9821 Containment minipurge inlet 5
18 HV9823 Containment minipurge inlet 5
19 HV9824 Containment minipurge outlet 5
19 HV9825 Containment minipurge outlet 5

to include all CIAS-actuated valves

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TABLE 3.6-1
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
A. CONTAINMENT ISOLATION (CIAS)			
1	HV-0510	Pressurizer steam space sample	25 ~~10~~
1	HV-0511	Pressurizer steam space sample	10
2	TV-9267	Letdown line to letdown heat exchanger	~~5~~ 10
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	25 ~~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
~~[illegible]~~	~~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
30~~A~~ B	HV-7801	Containment air radioactivity monitor outlet	1

30 B

per FSAR T 6.2-30 revision

per FSAR T 6.2-30 revision

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JUL 07 1981

6/30/81

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TABLE 3.6-1 (Continued)
~~(Continued)~~

CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM 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	15 ~~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	25 ~~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

Remote manual valves; do not receive CIAS

B. CONTAINMENT PURGE (CPIS)

18	HV-9949	Containment purge inlet (normal)	12 ~~10~~
18	HV-9948	Containment purge inlet (normal)	~~10~~ 12
18	HV-9821	Containment mini-purge inlet	5
18	HV-9823	Containment mini-purge inlet	5
19	HV-9950	Containment purge outlet (normal)	12 ~~10~~
19	HV-9951	Containment purge outlet (normal)	12 ~~10~~
19	HV-9824	Containment mini-purge outlet	5
19	HV-9825	Containment mini-purge outlet	5

updated per table

JUL 07 1981

TABLE 3.6-1 (Continued)
~~(Continued)~~

CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM ISOLATION TIME (SEC)
C. MANUAL			
6	2"-099-C-376 *	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

per FSAR revision

Necessary to permit SIT sampling untill first refueling

Closed system returning to RCS; Type C not required

move to Section D

Note Not req'd by GDC only provided for Type C testing of penetration

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TABLE 3.6-1 (Continued)
~~(Continued)~~
CONTAINMENT ISOLATION VALVES

Necessary to permit SIT sampling until first refueling

Closed system returning to RCS, Type C not required

move to Section D

Note, Not req'd by GDC only provided for Type C testing of penetration

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM 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

SAN ONOFRE-UNIT 2

9 PSV9349 Shutdown cooling relief NA

TABLE 3.6-1 (Continued)

~~(Continued)~~

CONTAINMENT ISOLATION VALVES

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PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM 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

42 HV 6223 Component Cooling water inlet N/A
43 HV 6236 Component Cooling water outlet N/A

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TABLE 3.6-1 (Continued)
~~(Continued)~~
CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM 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

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TABLE 3.6-1 ~~(Continued)~~ (Continued)

CONTAINMENT ISOLATION VALVES

PENETRATION NUMBER	VALVE NUMBER	FUNCTION	MAXIMUM 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

check this

*May be opened on an intermittent basis under administrative control.

#Not subject to Type C leakage tests.

@ Shutdown Cooling valves may be opened in Mode 4

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CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

DRAFT

LIMITING CONDITION FOR OPERATION

3.6.4.1 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

4.6.4.1 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.

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ELECTRIC HYDROGEN RECOMBINERS - W

DRAFT

LIMITING CONDITION FOR OPERATION

3.6.4.2 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

4.6.4.2 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.

b. At least once per 18 months by:

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

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

3. Verifying the integrity of the heater electrical circuits by performing a resistance to ground test following the above required functional test. The resistance to ground for any heater phase shall be greater than or equal to 10,000 ohms.

continuity test and a

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CONTAINMENT DOME AIR CIRCULATORS

LIMITING CONDITION FOR OPERATION

3.6.4.3 Two independent dome air circulator trains shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With one dome air circulator train inoperable, restore the inoperable train to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.3 Each dome air circulator train shall be demonstrated OPERABLE:

a. At least once per 18 months by starting each train on a CCAS Test 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.

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CONTAINMENT SYSTEMS

HYDROGEN MONITORS

LIMITING CONDITION FOR OPERATION

3.6.~~5~~4.4 Two independent containment hydrogen monitors shall be OPERABLE.

APPLICABILITY: MODES 1 and 2

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.~~7~~4.4 Each hydrogen monitor shall be demonstrated OPERABLE by the 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.

H_2 monitoring is performed by the H_2 analyzer at SO2/3

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

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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. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER		LIFT SETTING ($\pm$ 1%)*	ORIFICE SIZE
	Line No. 1	Line No. 2		
a.	2PSV-8401	2PSV-8410	1100 psia	16 in^2
b.	2PSV-8402	2PSV-8411	1107 psia	16 in^2
c.	2PSV-8403	2PSV-8412	1114 psia	16 in^2
d.	2PSV-8404	2PSV-8413	1121 psia	16 in^2
e.	2PSV-8405	2PSV-8414	1128 psia	16 in^2
f.	2PSV-8406	2PSV-8415	1135 psia	16 in^2
g.	2PSV-8407	2PSV-8416	1142 psia	16 in^2
h.	2PSV-8408	2PSV-8417	1149 psia	16 in^2
i.	2PSV-8409	2PSV-8418	1155 psia	16 in^2

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

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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	90
2	79
3	68
4	57
5	45
6	34
7	23
8	0

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AUXILIARY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

a. ~~One~~ Two feedwater pumps capable of being powered from ~~an~~ separate OPERABLE emergency ~~bus~~ busses, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(two capable of being powered from separate OPERABLE emergency busses and one capable of being powered from an OPERABLE steam supply)

a. With one auxiliary feedwater pump inoperable, restore ~~the required~~ at least three 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, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.~~

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. ~~At least once per 31 days by:~~ By ~~testing pursuant~~ to Specification 4.0.5.

~~1. Testing the motor driven pump pursuant to Specification 4.0.5.~~

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

b. ~~3.~~ At least once per 31 days by verifying ~~Verifying~~ that each valve ~~in~~ (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

Note: Stm-driven pump is fully safety-related (IE electrical power (DC) plus Seismic I steam supply)

PLANT SYSTEMS DRAFT

AUXILIARY FEEDWATER SYSTEM

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LIMITING CONDITION FOR OPERATION

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

a. ~~One~~ Two feedwater pump~~s~~ capable of being powered from ~~an~~ separate OPERABLE emergency ~~bus~~ busses, and X X

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(two capable of being powered from separate OPERABLE emergency busses and one capable of being powered from an OPERABLE steam supply)

a. With one auxiliary feedwater pump inoperable, restore ~~the required~~ at least three 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. X X

~~b. With two auxiliary feedwater pumps inoperable, immediately initiate corrective action to restore at least one auxiliary feedwater pump to OPERABLE status as soon as possible.~~ X

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. ~~At least once per 31 days by:~~ X
By testing pursuant to Specification 4.0.5.

~~1. Testing the motor driven pump pursuant to Specification 4.0.5.~~

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

Note: Stm driven pump is fully safety-related (~~II~~ electrical power (DC) plus seismic steam supply)

b. ~~3.~~ At least once per 31 days by verifying that each NON-AUTOMATIC valve in ~~Verifying that each valve in (manual, power operated or automatic)~~ the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position. X

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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 upon receipt of an EFAS test signal.
2. Verifying that each pump starts automatically upon receipt of an EFAS test signal.

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CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CSTs) ~~(CST)~~ shall be OPERABLE with a contained volume of at least ~~(160,000)~~ 144,000 gallons, in T121 and 180,000 gallons in T120

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tanks inoperable, within 4 hours either:

a. Restore the CSTs 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.~~

AFW pumps draw suction only from T-121; T-120 drained into T-121 but not valved direct to pumps

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.~~

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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.

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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.

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TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY
SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross 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.

PLANT SYSTEMS

MAIN STEAM LINE ISOLATION VALVES

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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 5.0 seconds when tested pursuant to Specification 4.0.5.

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3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperatures of both the primary and secondary coolants in the steam generators shall be greater than 70°F when the pressure of either coolant in the steam generator is greater than 200 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

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

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

SURVEILLANCE REQUIREMENTS

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

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3/4.7.3 COMPONENT COOLING WATER SYSTEM

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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.

3/4.7.4 SALT WATER COOLING SYSTEM

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LIMITING CONDITION FOR OPERATION

3.7.4 At least two independent salt water cooling loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one salt 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 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 an SIAS test signal.

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3/4.7.5 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.5 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

4.7.5 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 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 10 hours with the heaters on automatic. ×

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:

air conditioning unit, and 1000 cfm ± 10% ventilation unit

1. Verifying that with the system operating at a flow rate of 35485 cfm ± 10% and exhausting through the respective 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% air conditioning unit and 1% ventilation unit when the system is tested by admitting cold DOP at the ~~system~~ respective intake.

to reflect SO 2/3 system configuration

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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% for the ventilation unit and 35,485 cfm ± 10% for the 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% for the ventilation unit and 35,485 cfm ± 10% for the 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)~~ 4.3 inches Water Gauge ventilation unit and less than 7.3 inches Water Gauge air conditioning unit while operating the system at a flow rate of 1000 cfm ± 10% for the ventilation unit and 35,485 cfm ± 10% for the air conditioning unit.

2. Verifying that on a control room isolation test signal, the system automatically switches into ~~a recirculation~~ the emergency mode of operation with flow through the HEPA filters and charcoal adsorber banks.

4.~~3~~. Verifying that the system maintains the control room at a positive pressure of greater than or equal to ~~(1/4)~~ 1/8 inch W.G. relative to the outside atmosphere during system operation in the emergency mode.

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X

5.~~4~~. Verifying that the heaters dissipate 3.2 kw ± 5% when tested in accordance with ANSI N510-1975.

3. Verifying that on a toxic gas isolation test signal, the system automatically switches into the isolation mode of operation with flow through the HEPA filters and charcoal ~~adsorber~~ banks.

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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[illegible]%~~ 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% for the ventilation unit and 35,485 cfm ± 10% for the 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% for the ventilation unit and 35,485 cfm ± 10% for the air conditioning unit.

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3/4.7.6 ECCS PUMP ROOM EXHAUST AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.6 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

4.7.6 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.)

No such system at SO 2/3

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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.)

No such system at SO 2/3

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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%.

no such system at SO 2/3

*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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PLANT SYSTEMS

3/4.7.~~8~~6 SNUBBERS

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.~~8~~6.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 performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7-4a and 3.7-4b. ~~If less then two 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.

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SURVEILLANCE REQUIREMENTS (Continued)

b. Visual Inspection Acceptance Criteria

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) in those locations where snubber movement can be manually induced without disconnecting the snubber, that 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.d or 4.7.9.e, as applicable. However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be determined inoperable and cannot be determined OPERABLE via functional testing for the purpose of establishing the next visual inspection interval. 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* of either (1) at least 10% of the total of each type of snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria of Specification 4.7.8.d or 4.7.8.e, an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers have been functionally tested.

*The requirements of this section for functionally testing mechanical snubbers may be waived until startup following the first refueling outage.

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SURVEILLANCE REQUIREMENTS (Continued)

or (2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.1, which includes acceptance and rejection criteria. On Figure 4.7.1, "C" is the total number of snubbers found with locking velocity or bleed rate or drag force (if applied) not meeting the acceptance requirements. The cumulative number of snubbers tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.1. If at any time the point plotted falls in the "Reject" region all snubbers of that type shall be functionally tested. If at any time the point plotted falls in the "Accept" region testing of that type of snubber shall be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. The representative sample shall be selected randomly from the total population identified in Tables 3.7-4a and 3.7-4b.

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 placed in the same location as snubbers which failed the previous functional test shall be retested during the next test period. Test results of these snubbers shall not be included in the sampling plan.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

*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.

SURVEILLANCE REQUIREMENTS (Continued)

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. Drag force shall not have increased more than 50 percent since the last surveillance test.

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.

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.

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SURVEILLANCE REQUIREMENTS (Continued)

Concurrent with the first inservice visual inspection and at least once per 18 months thereafter, the installation and mainenance 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 by more than 10% prior to the next scheduled snubber service life review. If the indicated service life will be exceeded by more than 10% 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. The results of the reevaluation may be used to justify a change to the service life of the snubber. This reevaluation, replacement or reconditioning shall be indicated in the records.

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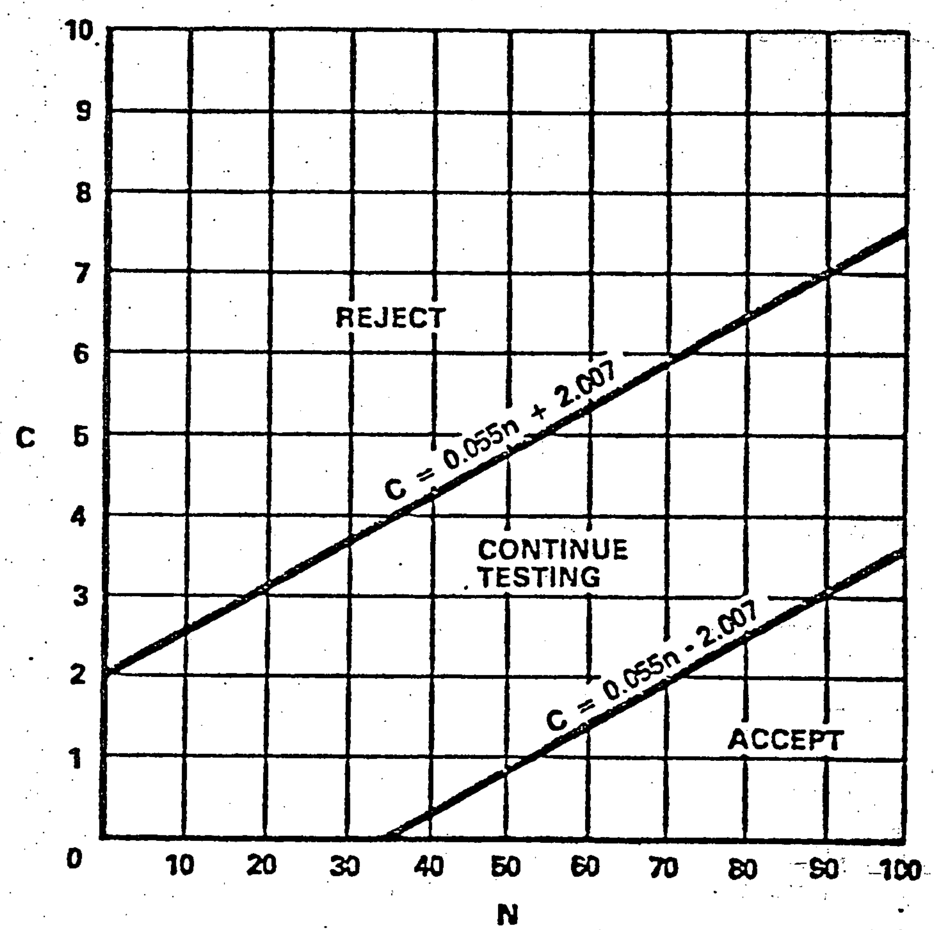

FIGURE 4.7.1 SAMPLING PLAN FOR SUNBBER FUNCTIONAL TEST

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

Radiation Zone Key for Table 3.7-4b

A - High radiation zone (>100 MR/HR)

B - Low radiation zone during normal operation. Higher radiation zone during shutdown.

C - Low radiation zone (<100 MR/HR)

D - Consider plant operation and/or special draining procedures for access.

TABLE 3.7-4a

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)
1	Reactor Coolant Pump 51' Elev	I	Yes	Yes
2	Reactor Coolant Pump 51' Elev	I	Yes	Yes
3	Reactor Coolant Pump 51' Elev	I	Yes	Yes
4	Reactor Coolant Pump 51' Elev	I	Yes	Yes
5	Steam Generator Upper Support 71' Elev	I	Yes	Yes
6	Steam Generator Upper Support 71' Elev	I	Yes	Yes
7	Steam Generator Upper Support 71' Elev	I	Yes	Yes
8	Steam Generator Upper Support 71' Elev	I	Yes	Yes

*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.

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
1	S2-RC-042-H-00C	2C06	2BBA	A		
2	S2-RC-140-H-00M	2C09	2BBA	A		
3	S2-RC-149-H-00A	2C09	2BBA	A		
4	S2-RC-149-H-00C	2C09	2BBA	A		
5	S2-RC-150-H-00B	2C10	2BBA	A		
6	S2-RC-157-H-00L	2C09	2BBA	A		
7	S2-RC-017-H-00A	2C05	2BHA	B		
8	S2-RC-017-H-00B	2C05	2BHA	B		
9	S3-ST-016-H-00C	2C05	2BMD	C		
10	S3-ST-001-H-00F	3C06	3ABB	C		
11	S3-BM-028-H-0AM	CA03	3BGB	D		
12	S3-CC-016-H-00R	3C01	3EGA	B		
13	S3-CC-016-H-00Y	3C01	3EGA	B		
14	S3-SS-030-H-00M	3C01	3EGA	B		
15	S2-SS-030-H-003	2C05	2SJA	A		
16	S2-SS-030-H-005	2C09	2SJA	A		
17	S2-SS-030-H-007	2C09	2SJA	A		
18	S2-SS-030-H-013	2C09	2SJA	A		
19	S2-SS-030-H-021	2C09	2SJA	A		
20	S2-SS-030-H-035	2C09	2SJA	A		
21	S2-SS-030-H-039	2C09	2SJA	A		
22	S2-SS-030-H-045	2C09	2SJA	A		
23	S2-SS-030-H-008	2C09	2SJA	B		
24	S2-SS-030-H-006	2C09	2SJA	C		
25	S2-SS-030-H-008	2C09	2SJA	C		
26	S2-SS-030-H-016	2C09	2SJA	A		
27	S2-SS-030-H-019	2C09	2SJA	C		
28	S2-SS-030-H-023	2C09	2SJA	C		
29	S2-SS-030-H-030	2C09	2SJA	C		
30	S2-SS-037-H-010	2C09	2SJA	C		
31	S2-SS-037-H-010	2C09	2SJA	C		
32	S2-SS-037-H-011	2C09	2SJA	C		
33	S2-SS-042-H-004	2C09	2SJA	C		
34	S2-SS-042-H-010	2C09	2SJA	C		
35	S2-SS-042-H-018	2C09	2SJA	C		
36	S2-SS-042-H-021	2C09	2SJA	A		
37	S2-SS-043-H-003	2C09	2SJA	C		

*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.

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
38	S2-SS-167-H-001	2C10	2SJA	A		
39	S2-SS-167-H-003	2C10	2SJA	A		
40	S2-SS-167-H-005	2C06	2SJA	A		
41	S2-SS-167-H-009	2C06	2SJA	A		
42	S2-SS-167-H-022	2C05	2SJA	A		
43	S2-SS-167-H-043	2C09	2SJA	A		
44	S2-SS-167-H-047	2C09	2SJA	C		
45	S2-SS-167-H-062	2C05	2SJA	A		
46	S2-SS-167-H-064	2C05	2SJA	A		
47	S2-SS-167-H-069	2C05	2SJA	A		
48	S2-SS-167-H-074	2C05	2SJA	A		
49	S2-SS-168-H-008	2C09	2SJA	C		
50	S2-SS-037-H-006	2C09	2SJA	B		
51	S2-SS-043-H-004	2C09	2SJA	B		
52	S2-SI-045-H-028	2C05	2SJA	C		
53	S2-ST-001-H-014	2C02	SABA	C		
54	S2-ST-001-H-021	2C02	2ABA	C		
55	S2-ST-001-H-022	2C02	2ABA	C		
56	S2-ST-001-H-023	2C02	2ABA	C		
57	S2-ST-001-H-031	2C02	2ABA	C		
58	S2-ST-001-H-034	2C02	2ABA	C		
59	S2-ST-002-H-014	2C01	2ABA	C		
60	S2-ST-002-H-026	2C01	2ABA	C		
61	S2-ST-002-H-034	2C01	2ABA	C		
62	S2-ST-017-H-004	2C06	2ABA	C		
63	S2-ST-017-H-005	2C06	2ABA	C		
64	S2-ST-017-H-008	2C06	2ABA	C		
65	S2-ST-017-H-009	2C06	2ABA	C		
66	S2-ST-577-H-014	2C01	2ABA	C		
67	S2-ST-577-H-018	2C02	2ABA	C		
68	S2-ST-577-H-021	2C02	2ABA	C		
69	S2-ST-577-H-032	2C02	2ABA	C		
70	S2-FW-189-H-010	2C06	2ABB	C		
71	S2-FW-189-H-012	2C06	2ABB	C		
72	S2-FW-189-H-013	2C06	2ABB	C		
73	S2-FW-189-H-014	2C06	2ABB	C		
74	S2-FW-189-H-017	2C06	2ABB	C		
75	S2-FW-190-H-011	2C05	2ABB	C		
76	S2-FW-190-H-013	2C05	2ABB	C		
77	S2-FW-190-H-014	2C05	2ABB	C		
78	S2-FW-190-H-015	2C05	2ABB	C		
79	S2-FW-190-H-018	2C05	2ABB	C		
80	S2-ST-001-H-001	2C06	2ABB	C		
81	S2-ST-001-H-002	2C06	2ABB	C		
82	S2-ST-001-H-004	2C06	2ABB	C		
83	S2-ST-001-H-005	2C06	2ABB	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
84	S2-ST-001-H-007	2C06	2ABB	C		
85	S2-ST-001-H-008	2C06	2ABB	C		
86	S2-ST-001-H-009	2C06	2ABB	C		
87	S2-ST-001-H-025	2C06	2ABB	C		
88	S2-ST-001-H-040	2C02	2ABB	C		
89	S2-ST-001-H-042	2C06	2ABB	C		
90	S2-ST-001-H-044	2C02	2ABB	C		
91	S2-ST-001-H-048	2C02	2ABB	C		
92	S2-ST-001-H-049	2C02	2ABB	C		
93	S2-ST-002-H-001	2C05	2ABB	C		
94	S2-ST-002-H-002	2C05	2ABB	C		
95	S2-ST-002-H-004	2C05	2ABB	C		
96	S2-ST-002-H-005	2C05	2ABB	C		
97	S2-ST-002-H-007	2C05	2ABB	C		
98	S2-ST-002-H-008	2C05	2ABB	C		
99	S2-ST-002-H-009	2C05	2ABB	C		
100	S2-ST-002-H-022	2C05	2ABB	C		
101	S2-ST-002-H-040	2C01	2ABB	C		
102	S2-ST-002-H-041	2C01	2ABB	C		
103	S2-ST-002-H-042	2C01	2ABB	C		
104	S2-ST-002-H-046	2C01	2ABB	C		
105	S2-ST-022-H-047	2C01	2ABB	C		
106	S2-ST-004-H-001	2C01	2ABB	C		
107	S2-ST-004-H-003	2C01	2ABB	C		
108	S2-ST-014-H-011	2C10	2ABB	C		
109	S2-ST-014-H-014	2C10	2ABB	C		
110	S2-ST-014-H-022	2C06	2ABB	A		
111	S2-ST-014-H-023	2C06	2ABB	A		
112	S2-ST-014-H-039	2C10	2ABB	C		
113	S2-ST-014-H-041	2C06	2ABB	C		
114	S2-ST-014-H-042	2C06	2ABB	C		
115	S2-ST-014-H-051	2C10	2ABB	C		
116	S2-ST-014-H-052	2C10	2ABB	C		
117	S2-ST-015-H-016	2C06	2ABB	A		
118	S2-ST-015-H-017	2C06	2ABB	A		
119	S2-ST-015-H-018	2C06	2ABB	A		
120	S2-ST-015-H-019	2C06	2ABB	A		
121	S2-ST-016-H-002	2C05	2ABB	A		
122	S2-ST-016-H-003	2C09	2ABB	A		
123	S2-ST-016-H-031	2C09	2ABB	A		
124	S2-ST-018-H-002	2C09	2ABB	C		
125	S2-ST-018-H-006	2C09	2ABB	C		
126	S2-ST-018-H-007	2C09	2ABB	C		
127	S2-ST-018-H-009	2C09	2ABB	C		
128	S2-ST-018-H-020	2C09	2ABB	A		
129	S2-ST-018-H-033	2C09	2ABB	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
130	S2-ST-018-H-035	2C09	2ABB	A		
131	S2-ST-018-H-036	2C09	2ABB	A		
132	S2-ST-018-H-039	2C09	2ABB	A		
133	S2-ST-309-H-003	2C01	2ABB	C		
134	S2-ST-578-H-001	2C02	2ABB	C		
135	S2-ST-578-H-003	2C02	2ABB	C		
136	S2-ST-580-H-002	2C02	2ABB	C		
137	S2-ST-583-H-003	2C01	2ABB	C		
138	S2-FW-222-H-004	2C06	2ALA	C		
139	S2-FW-222-H-008	2C10	2ALA	C		
140	S2-FW-222-H-010	2C06	2ALA	C		
141	S2-FW-223-H-002	2C05	2ALA	C		
142	S2-F2-223-H-003	2C05	2ALA	C		
143	S2-FW-223-H-007	2C05	2ALA	C		
144	S2-FW-223-H-012	2C05	2ALA	C		
145	S2-FW-301-H-003	2C12	2ANA	C		
146	S2-FW-301-H-005	2C12	2ANA	C		
147	S2-FW-301-H-008	2C12	2ANA	C		
148	S2-FW-302-H-003	2C12	2ANA	C		
149	S2-CC-070-H-004	2C05	2BBA	A		
150	S2-CC-072-H-005	2C05	2BBA	A		
151	S2-CC-072-H-006	2C05	2BBA	A		
152	S2-CC-074-H-004	2C05	2BBA	A		
153	S2-CC-074-H-008	2C05	2BBA	A		
154	S2-CC-074-H-010	2C09	2BBA	A		
155	S2-CC-076-H-006	2C05	2BBA	A		
156	S2-CC-076-H-009	2C05	2BBA	A		
157	S2-CC-076-H-012	2C09	2BBA	A		
158	S2-CC-078-H-006	2C06	2BBA	A		
159	S2-CC-078-H-008	2C10	2BBA	A		
160	S2-CC-078-H-010	2C10	2BBA	A		
161	S2-CC-080-H-007	2C06	2BBA	A		
162	S2-CC-080-H-008	2C10	2BBA	A		
163	S2-CC-080-H-010	2C10	2BBA	A		
164	S2-CC-082-H-007	2C06	2BBA	A		
165	S2-CC-082-H-008	2C06	2BBA	A		
166	S2-CC-082-H-009	2C06	2BBA	A		
167	S2-CC-082-H-011	2C10	2BBA	A		
168	S2-CC-082-H-012	2C06	2BBA	A		
169	S2-CC-082-H-013	2C06	2BBA	A		
170	S2-CC-084-H-004	2C06	2BBA	A		
171	S2-CC-084-H-005	2C06	2BBA	A		
172	S2-CC-084-H-006	2C06	2BBA	A		
173	S2-CC-084-H-008	2C06	2BBA	A		
174	S2-CC-084-H-009	2C06	2BBA	A		
175	S2-CC-084-H-010	2C06	2BBA	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
176	S2-CC-364-H-002	2C09	2BBA	A		
177	S2-CC-365-H-003	2C09	2BBA	A		
178	S2-CC-366-H-003	2C05	2BBA	A		
179	S2-CC-367-H-002	2C06	2BBA	A		
180	S2-CC-368-H-001	2009	2BBA	A		
181	S2-CC-371-H-001	2C06	2BBA	A		
182	S2-CC-372-H-001	2C09	2BBA	A		
183	S2-CC-372-H-003	2C09	2BBA	A		
184	S2-CC-373-H-002	2C10	2BBA	A		
185	S2-CC-374-H-001	2C05	2BBA	A		
186	S2-CC-375-H-002	2C06	2BBA	A		
187	S2-CC-375-H-003	2C06	2BBA	A		
188	S2-RC-018-H-001	2C05	2BBA	A		
189	S2-RC-021-H-001	2C05	2BBA	A		
190	S2-RC-021-H-002	2C05	2BBA	A		
191	S2-RC-022-H-002	2C05	2BBA	A		
192	S2-RC-022-H-004	2C05	2BBA	A		
193	S2-RC-022-H-010	2C05	2BBA	A		
194	S2-RC-031-H-001	2C10	2BBA	A		
195	S2-RC-052-H-002	2C09	2BBA	B		
196	S2-RC-011-H-001	2C09	2BBA	A		
197	S2-RC-012-H-026	2C09	2BBA	C		
198	S2-RC-012-H-029	2C09	2BBA	C		
199	S2-RC-013-H-002	2C05	2BBA	A		
200	S2-RC-015-H-017	2C05	2BBA	A		
201	S2-RC-015-H-018	2C05	2BBA	A		
202	S2-RC-015-H-020	2C09	2BBA	A		
203	S2-RC-016-H-004	2C06	2BBA	A		
204	S2-RC-016-H-008	2C05	2BBA	A		
205	S2-RC-016-H-009	2C06	2BBA	A		
206	S2-RC-016-H-010	2C06	2BBA	A		
207	S2-RC-023-H-001	2C05	2BBB	A		
208	S2-RC-032-H-002	2C09	2BBB	C		
209	S2-RC-033-H-022	2C09	2BBB	C		
210	S2-RC-034-H-012	2C09	2BBB	C		
211	S2-RC-034-H-014	2C09	2BBB	C		
212	S2-RC-034-H-015	2C09	2BBB	C		
213	S2-RC-034-H-018	2C09	2BBB	C		
214	S2-RC-034-H-021	2C09	2BBB	C		
215	S2-RC-034-H-023	2C09	2BBB	C		
216	S2-RC-049-H-002	2C09	2BBB	A		
217	S2-RC-060-H-003	2C09	2BBB	A		
218	S2-RC-060-H-009	2C09	2BBB	A		
219	S2-RC-060-H-012	2C09	2BBB	A		
220	S2-RC-061-H-012	2C10	2BBB	A		
221	S2-RC-061-H-013	2C06	2BBB	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
222	S2-RC-061-H-014	2C06	2BBB	A		
223	S2-RC-061-H-016	2C06	2BBB	A		
224	S2-RC-062-H-003	2C09	2BBB	A		
225	S2-RC-062-H-009	2C09	2BBB	A		
226	S2-RC-062-H-012	2C09	2BBB	A		
227	S2-RC-062-H-013	2C09	2BBB	A		
228	S2-RC-062-H-014	2C09	2BBB	C		
229	S2-RC-062-H-015	2C09	2BBB	C		
230	S2-RC-064-H-002	2C06	2BBB	A		
231	S2-RC-064-H-003	2C06	2BBB	A		
232	S2-RC-072-H-002	2C06	2BBB	C		
233	S2-RC-073-H-019	2C05	2BBB	A		
234	S2-RC-073-H-021	2C05	2BBB	A		
235	S2-RC-073-H-023	2C05	2BBB	A		
236	S2-RC-073-H-025	2C09	2BBB	A		
237	S2-RC-073-H-026	2C06	2BBB	A		
238	S2-RC-073-H-031	2C06	2BBB	C		
239	S2-RC-094-H-009	2C09	2BBB	C		
240	S2-RC-096-H-025	2C09	2BBB	C		
241	S2-RC-147-H-001	2C05	2BBB	A		
242	S2-SI-021-H-026	2C14	2BBB	B		
243	S2-SI-043-H-016	2C09	2BBB	C		
244	S2-SI-043-H-018	2C09	2BBB	A		
245	S2-SI-043-H-020	2C09	2BBB	A		
246	S2-SI-043-H-021	2C09	2BBB	A		
247	S2-SI-044-H-001	2C05	2BBB	C		
248	S2-SI-044-H-003	2C05	2BBB	C		
249	S2-SI-044-H-004	2C05	2BBB	C		
250	S2-SI-044-H-006	2C05	2BBB	A		
251	S2-SI-044-H-009	2C05	2BBB	A		
252	S2-SI-045-H-005	2C05	2BBB	C		
253	S2-SI-045-H-008	2C05	2BBB	C		
254	S2-SI-045-H-011	2C06	2BBB	C		
255	S2-SI-045-H-015	2C06	2BBB	A		
256	S2-SI-045-H-018	2C06	2BBB	A		
257	S2-SI-046-H-002	2C06	2BBB	A		
258	S2-SI-046-H-003	2C10	2BBB	C		
259	S2-SI-046-H-006	2C10	2BBB	C		
260	S2-SI-046-H-010	2C10	2BBB	A		
261	S2-SI-046-H-012	2C10	2BBB	A		
262	S2-SI-062-H-001	2C10	2BBB	A		
263	S2-SI-062-H-002	2C10	2BBB	A		
264	S2-SI-063-H-020	2C09	2BBB	A		
265	S2-SI-064-H-007	2C05	2BBB	A		
266	S2-SI-064-H-013	2C05	2BBB	A		
267	S2-SI-064-H-019	2C05	2BBB	C		

TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
268	S2-SI-064-H-020	2C05	2BBB	A		
269	S2-SI-065-H-030	2C05	2BBB	C		
270	S2-SI-065-H-002	2C06	2BBB	A		
271	S2-SI-065-H-003	2C06	2BBB	A		
272	S2-SI-065-H-004	2C06	2BBB	A		
273	S2-SI-065-H-005	2C06	2BBB	A		
274	S2-SI-065-H-009	2C06	2BBB	A		
275	S2-SI-065-H-010	2C06	2BBB	A		
276	S2-SI-065-H-014	2C06	2BBB	A		
277	S2-SI-065-H-017	2C06	2BBB	A		
278	S2-SI-065-H-019	2C06	2BBB	A		
279	S2-SI-065-H-021	2C06	2BBB	A		
280	S2-SI-065-H-024	2C06	2BBB	A		
281	S2-SI-066-H-005	2C10	2BBB	C		
282	S2-SI-066-H-010	2C10	2BBB	A		
283	S2-SI-066-H-013	2C10	2BBB	A		
284	S2-SI-066-H-015	2C10	2BBB	A		
285	S2-SI-066-H-018	2C10	2BBB	A		
286	S2-SI-066-H-022	2C10	2BBB	A		
287	S2-SI-066-H-024	2C10	2BBB	A		
288	S2-SI-066-H-025	2C10	2BBB	A		
289	S2-SI-066-H-026	2C10	2BBB	A		
290	S2-SI-066-H-028	2C10	2BBB	A		
291	S2-SI-066-H-030	2C10	2BBB	A		
292	S2-SI-087-H-007	2C05	2BBB	C		
293	S2-SI-087-H-011	2C05	2BBB	C		
294	S2-SI-087-H-019	2C06	2BBB	C		
295	S2-SI-087-H-021	2C06	2BBB	C		
296	S2-SI-087-H-022	2C06	2BBB	C		
297	S2-SI-087-H-025	2C06	2BBB	C		
298	S2-SI-087-H-026	2C06	2BBB	C		
299	S2-SI-152-H-001	2C09	2BBB	C		
300	S2-SI-155-H-001	2C10	2BBB	C		
301	S2-SI-165-H-006	2C09	2BBB	A		
302	S2-SI-165-H-007	2C09	2BBB	A		
303	S2-SI-166-H-002	2C09	2BBB	C		
304	S2-SI-168-H-001	2C06	2BBB	C		
305	S2-SI-168-H-002	2C06	2BBB	C		
306	S2-SI-168-H-003	2C06	2BBB	C		
307	S2-RC-034-H-010	2C09	2BBB	C		
308	S2-RC-061-H-010	2C06	2BBB	A		
309	S2-SI-045-H-013	2C06	2BBB	C		
310	S2-SI-063-H-022	2C09	2BBB	A		
311	S2-SI-063-H-023	2C09	2BBB	A		
312	S2-BM-021-H-003	CA03	2BGA	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
313	S2-BM-022-H-010	CA03	2BGA	D		
314	S2-BM-022-H-011	CA03	2BGA	D		
315	S2-BM-022-H-012	CA03	2BGA	D		
316	S2-BM-022-H-014	CA03	2BGA	D		
317	S2-BM-031-H-008	CA03	2BGA	D		
318	S2-BM-031-H-018	CA03	2BGA	D		
319	S2-BM-031-H-019	CA03	2BGA	D		
320	S2-BM-036-H-003	CA03	2BGA	D		
321	S2-BM-036-H-005	CA03	2GBA	D		
322	S2-BM-035-H-007	CA03	2BGA	C		
323	S2-FS-057-H-015	CA03	2BGA	D		
324	S2-FS-057-H-059	2C05	2BGA	C		
325	S2-FS-057-H-061	2C05	2BGA	C		
326	S2-FS-057-H-068	2C09	2BGA	C		
327	S2-FS-057-H-070	2C09	2BGA	C		
328	S2-FS-057-H-075	CA03	2BGA	C		
329	S2-FS-057-H-078	CA03	2BGA	C		
330	S2-FS-057-H-079	2C09	2BGA	C		
331	S2-FS-057-H-080	2C09	2BGA	C		
332	S2-FS-057-H-082	2C09	2BGA	C		
333	S2-FS-057-H-083	2C09	2BGA	C		
334	S2-FS-057-H-085	2C09	2BGA	C		
335	S2-FS-057-H-086	2C09	2BGA	D		
336	S2-FS-057-H-087	2C09	2BGA	C		
337	S2-FS-057-H-098	2C01	2BGA	C		
338	S2-RC-022-H-009	2C05	2BGA	A		
339	S2-VC-001-H-008	2C05	2BGA	A		
340	S2-VC-001-H-009	2C05	2BGA	A		
341	S2-VC-001-H-010	2C05	2BGA	A		
342	S2-VC-001-H-013	2C05	2BGA	A		
343	S2-VC-001-H-014	2C05	2BGA	A		
344	S2-VC-001-H-015	2C05	2BGA	A		
345	S2-VC-001-H-016	2C05	2BGA	A		
346	S2-VC-003-H-019	2C09	2BGA	B		
347	S2-VC-004-H-017	CA05	2BGA	D		
348	S2-VC-007-H-005	CA03	2BGA	A		
349	S2-VC-007-H-017	CA03	2BGA	A		
350	S2-VC-007-H-018	CA03	2BGA	D		
351	S2-VC-007-H-025	CA03	2BGA	D		
352	S2-VC-008-H-039	2C09	2BGA	B		
353	S2-VC-009-H-010	2C09	2BGA	A		
354	S2-VC-009-H-011	2C09	2BGA	A		
355	S2-VC-009-H-012	2C09	2BGA	C		
356	S2-VC-010-H-006	2C05	2BGA	A		
357	S2-VC-010-H-014	2C09	2BGA	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
358	S2-VC-010-H-020	2C06	2BGA	A		
359	S2-VC-010-H-022	2C06	2BGA	A		
360	S2-VC-010-H-027	2C06	2BGA	A		
361	S2-VC-010-H-029	2C06	2BGA	C		
362	S2-VC-010-H-031	2C05	2BGA	C		
363	S2-VC-011-H-011	2C09	2BGA	C		
364	S2-VC-014-H-002	CA03	2BGA	D		
365	S2-VC-035-H-006	CA03	2BGA	D		
366	S2-VC-046-H-001	2C09	2BGA	C		
367	S2-VC-055-H-015	CA03	2BGA	D		
368	S2-VC-055-H-017	CA03	2BGA	D		
369	S2-VC-055-H-020	CA03	2BGA	D		
370	S2-VC-055-H-021	CA03	2BGA	D		
371	S2-VC-055-H-022	CA03	2BGA	D		
372	S2-VC-055-H-023	CA03	2BGA	D		
373	S2-VC-056-H-012	CA03	2BGA	D		
374	S2-VC-086-H-002	CA03	2BGA	D		
375	S2-VC-099-H-001	CA03	2BGA	D		
376	S2-VC-104-H-009	2C09	2BGA	C		
377	S2-VC-104-H-013	2C09	2BGA	A		
378	S2-VC-104-H-014	2C09	2BGA	A		
379	S2-VC-104-H-019	2C09	2BGA	B		
380	S2-RC-022-H-008	2C05	2BGA	A		
381	S2-VC-003-H-017	CA05	2BGA	A		
382	S2-VC-004-H-002	CA03	2BGA	A		
383	S2-VC-004-H-004	CA03	2BGA	A		
384	S2-VC-004-H-006	CA03	2BGA	A		
385	S2-VC-004-H-014	CA05	2BGA	A		
386	S2-VC-008-H-004	CA03	2BGA	D		
387	S2-VC-008-H-007	CA03	2BGA	D		
388	S2-VC-008-H-011	CA03	2BGA	D		
389	S2-VC-008-H-029	CA05	2BGA	A		
390	S2-VC-010-H-013	2C09	2BGA	A		
391	S2-VC-010-H-032	2C05	2BGA	A		
392	S2-VC-012-H-007	CA05	2BGA	A		
393	S2-VC-015-H-007	CA05	2BGA	D		
394	S2-VC-015-H-009	CA05	2BGA	A		
395	S2-VC-015-H-011	CA05	2BGA	A		
396	S2-VC-015-H-012	CA05	2BGA	A		
397	S2-VC-047-H-004	CA03	2BGA	D		
398	S2-VC-058-H-004	CA03	2BGA	D		
399	S2-VC-059-H-005	CA03	2BGA	D		
400	S2-BM-003-H-008	CA03	2BGB	D		
401	S2-BM-004-H-007	CA03	2BGB	D		
402	S2-BM-004-H-009	CA03	2BGB	D		
403	S2-BM-024-H-006	CA03	2BGB	D		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
404	S2-BM-025-H-008	CA03	2BGB	D		
405	S2-BM-037-H-004	CA03	2BGB	C		
406	S2-BM-037-H-011	CA03	2BGB	C		
407	S2-BM-037-H-017	CA03	2BGB	C		
408	S2-BM-037-H-019	CA03	2BGB	C		
409	S2-BM-037-H-020	CA03	2BGB	C		
410	S2-BM-037-H-022	CA03	2BGB	C		
411	S2-BM-056-H-002	CA03	2BGB	D		
412	S2-BM-057-H-002	CA03	2BGB	D		
413	S2-BM-057-H-003	CA03	2BGB	D		
414	S2-BM-057-H-005	CA03	2BGB	D		
415	S2-BM-057-H-006	CA03	2BGB	D		
416	S2-BM-057-H-009	CA05	2BGB	D		
417	S2-VC-116-H-007	CA03	2BGA	D		
418	S2-CS-019-H-009	2C01	2BHA	C		
419	S2-CS-020-H-001	2C02	2BHA	C		
420	S2-CS-020-H-003	2C02	2BHA	C		
421	S2-RC-016-H-003	2C06	2BHA	C		
422	S2-RC-017-H-002	2C06	2BHA	C		
423	S2-RC-017-H-030	2C06	2BHA	C		
424	S2-RC-017-H-033	2C06	2BHA	C		
425	S2-RC-017-H-036	2C05	2BHA	C		
426	S2-RC-017-H-038	2C05	2BHA	C		
427	S2-RC-017-H-040	2C05	2BHA	C		
428	S2-RC-017-H-044	2C01	2BHA	B		
429	S2-RC-017-H-050	2C05	2BHA	B		
430	S2-RC-017-H-052	2C05	2BHA	B		
431	S2-RC-017-H-054	2C05	2BHA	B		
432	S2-RC-017-H-055	2C05	2BHA	B		
433	S2-RC-017-H-056	2C05	2BHA	B		
434	S2-RC-017-H-057	2C05	2BHA	B		
435	S2-RC-017-H-067	2C05	2BHA	B		
436	S2-RC-017-H-069	2C05	2BHA	B		
437	S2-RC-017-H-070	2C05	2BHA	B		
438	S2-RC-017-H-071	2C05	2BHA	B		
439	S2-RC-017-H-072	2C05	2BHA	B		
440	S2-RC-017-H-076	2C01	2BHA	B		
441	S2-RC-017-H-077	2C01	2BHA	B		
442	S2-RC-017-H-080	2C06	2BHA	C		
443	S2-RC-017-H-081	2C06	2BHA	C		
444	S2-RC-017-H-084	2C05	2BHA	B		
445	S2-RC-017-H-085	2C05	2BHA	B		
446	S2-RC-017-H-090	2C01	2BHA	B		
447	S2-RC-017-H-091	2C05	2BHA	B		
448	S2-RC-025-H-002	2C05	2BHA	B		
449	S2-RC-025-H-006	2C05	2BHA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
450	S2-RC-025-H-009	2C05	2BHA	B		
451	S2-RC-025-H-011	2C05	2BHA	B		
452	S2-RC-025-H-013	2C05	2BHA	B		
453	S2-RC-027-H-004	2C05	2BHA	B		
454	S2-RC-027-H-005	2C05	2BHA	B		
455	S2-RC-050-H-005	2C02	2BHA	B		
456	S2-RC-050-H-007	2C02	2BHA	B		
457	S2-RC-050-H-014	2C02	2BHA	B		
458	S2-RC-050-H-019	2C02	2BHA	B		
459	S2-RC-050-H-021	2C02	2BHA	B		
460	S2-RC-051-H-001	2C06	2BHA	C		
461	S2-RC-071-H-001	2C06	2BHA	C		
462	S2-RC-071-H-003	2C06	2BHA	C		
463	S2-RC-071-H-004	2C06	2BHA	C		
464	S2-RC-072-H-004	2C06	2BHA	C		
465	S2-RC-076-H-001	2C06	2BHA	C		
466	S2-RC-134-H-001	2C05	2BHA	B		
467	S2-RC-134-H-002	2C05	2BHA	B		
468	S2-RC-134-H-003	2C05	2BHA	B		
469	S2-RC-134-H-004	2C05	2BHA	B		
470	S2-RC-134-H-006	2C06	2BHA	B		
471	S2-RC-134-H-007	2C05	2BHA	B		
472	S2-SI-001-H-024	2C02	2BHA	B		
473	S2-SI-002-H-024	2C07	2BHA	C		
474	S2-SI-002-H-029	2C07	2BHA	C		
475	S2-SI-003-H-003	2C06	2BHA	C		
476	S2-SI-003-H-006	2C02	2BHA	C		
477	S2-SI-003-H-017	2C01	2BHA	B		
478	S2-SI-003-H-018	2C01	2BHA	B		
479	S2-SI-004-H-003	2C06	2BHA	C		
480	S2-SI-004-H-006	2C02	2BHA	C		
481	S2-SI-007-H-004	2C04	2BHA	C		
482	S2-SI-008-H-003	2C02	2BHA	B		
483	S2-SI-009-H-003	2C02	2BHA	B		
484	S2-SI-009-H-011	2C01	2BHA	B		
485	S2-SI-010-H-001	2C02	2BHA	B		
486	S2-SI-010-H-003	2C02	2BHA	B		
487	S2-SI-010-H-006	2C02	2BHA	B		
488	S2-SI-010-H-012	2C02	2BHA	B		
489	S2-SI-012-H-004	2C05	2BHA	B		
490	S2-SI-014-H-012	2C02	2BHA	B		
491	S2-SI-014-H-018	2C01	2BHA	B		
492	S2-SI-014-H-019	2C01	2BHA	B		
493	S2-SI-014-H-021	2C01	2BHA	B		
494	S2-SI-017-H-006	2C05	2BHA	B		
495	S2-SI-020-H-033	2C05	2BHA	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
496	S2-SI-020-H-034	2C05	2BHA	C		
497	S2-SI-020-H-037	2C05	2BHA	C		
498	S2-SI-020-H-041	2C09	2BHA	C		
499	S2-SI-020-H-053	2C09	2BHA	C		
500	S2-SI-020-H-059	2C14	2BHA	B		
501	S2-SI-020-H-062	2C14	2BHA	B		
502	S2-SI-020-H-065	2C13	2BHA	C		
503	S2-SI-020-H-069	2C13	2BHA	C		
504	S2-SI-020-H-071	2C14	2BHA	B		
505	S2-SI-021-H-002	2C05	2BHA	B		
506	S2-SI-021-H-011	2C09	2BHA	B		
507	S2-SI-021-H-014	2C09	2BHA	B		
508	S2-SI-021-H-017	2C09	2BHA	B		
509	S2-SI-021-H-021	2C09	2BHA	B		
510	S2-SI-021-H-023	2C09	2BHA	B		
511	S2-SI-021-H-031	2C09	2BHA	B		
512	S2-SI-032-H-001	2C01	2BHA	B		
513	S2-SI-032-H-004	2C01	2BHA	B		
514	S2-SI-033-H-007	2C02	2BHA	B		
515	S2-SI-034-H-003	2C01	2BHA	B		
516	S2-SI-034-H-011	2C01	2BHA	B		
517	S2-SI-035-H-008	2C02	2BHA	B		
518	S2-SI-036-H-003	2C05	2BHA	B		
519	S2-SI-036-H-005	2C05	2BHA	B		
520	S2-SI-036-H-007	2C05	2BHA	B		
521	S2-SI-036-H-010	2C05	2BHA	B		
522	S2-SI-037-H-002	2C09	2BHA	B		
523	S2-SI-038-H-006	2C02	2BHA	B		
524	S2-SI-038-H-012	2C02	2BHA	B		
525	S2-SI-038-H-013	2C02	2BHA	B		
526	S2-SI-038-H-015	2C05	2BHA	B		
527	S2-SI-038-H-021	2C02	2BHA	B		
528	S2-SI-038-H-030	2C02	2BHA	B		
529	S2-SI-038-H-039	2C01	2BHA	B		
530	S2-SI-039-H-041	2C01	2BHA	B		
531	S2-SI-039-H-044	2C01	2BHA	B		
532	S2-SI-039-H-060	2C02	2BHA	B		
533	S2-SI-039-H-061	2C02	2BHA	B		
534	S2-SI-039-H-062	2C02	2BHA	B		
535	S2-SI-039-H-067	2C02	2BHA	B		
536	S2-SI-039-H-068	2C02	2BHA	B		
537	S2-SI-039-H-072	2C01	2BHA	B		
538	S2-RC-017-H-048	2C01	2BHA	B		
539	S2-SI-038-H-076	2C09	2BHA	B		
540	S2-SI-038-H-078	2C09	2BHA	B		
541	S2-SI-038-H-081	2C09	2BHA	B		

TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
542	S2-SI-038-H-084	2C09	2BHA	B		
543	S2-SI-038-H-087	2C09	2BHA	B		
544	S2-SI-038-H-089	2C05	2BHA	B		
545	S2-SI-038-H-092	2C02	2BHA	B		
546	S2-SI-038-H-094	2C02	2BHA	B		
547	S2-SI-038-H-095	2C01	2BHA	B		
548	S2-SI-038-H-100	2C13	2BHA	B		
549	S2-SI-038-H-104	2C09	2BHA	B		
550	S2-SI-038-H-105	2C14	2BHA	B		
551	S2-SI-038-H-106	2C09	2BHA	B		
552	S2-SI-048-H-004	2C02	2BHA	B		
553	S2-SI-048-H-007	2C02	2BHA	B		
554	S2-SI-048-H-011	2C02	2BHA	B		
555	S2-SI-048-H-012	2C02	2BHA	B		
556	S2-SI-049-H-003	2C01	2BHA	B		
557	S2-SI-050-H-003	2C01	2BHA	B		
558	S2-SI-050-H-006	2C01	2BHA	B		
559	S2-SI-052-H-010	2C02	2BHA	B		
560	S2-SI-052-H-022	2C02	2BHA	B		
561	S2-SI-052-H-023	2C02	2BHA	B		
562	S2-SI-053-H-003	2C01	2BHA	B		
563	S2-SI-059-H-009	2C09	2BHA	C		
564	S2-SI-062-H-005	2C10	2BHA	C		
565	S2-SI-063-H-018	2C09	2BHA	C		
566	S2-SI-066-H-032	2C10	2BHA	C		
567	S2-SI-067-H-002	2C09	2BHA	C		
568	S2-SI-074-H-002	2C10	2BHA	C		
569	S2-SI-074-H-003	2C10	2BHA	C		
570	S2-SI-077-H-006	2C06	2BHA	C		
571	S2-SI-077-H-008	2C06	2BHA	C		
572	S2-SI-077-H-012	2C06	2BHA	C		
573	S2-SI-108-H-003	2C02	2BHA	B		
574	S2-SI-109-H-002	2C02	2BHA	B		
575	S2-SI-109-H-005	2C02	2BHA	B		
576	S2-SI-132-H-002	2C02	2BHA	B		
577	S2-SI-139-H-005	2C05	2BHA	B		
578	S2-SI-139-H-012	2C05	2BHA	B		
579	S2-SI-139-H-015	2C01	2BHA	B		
580	S2-SI-139-H-031	2C01	2BHA	B		
581	S2-SI-145-H-003	2C05	2BHA	B		
582	S2-SI-145-H-005	2C05	2BHA	B		
583	S2-SI-146-H-004	2C09	2BHA	B		
584	S2-SI-149-H-010	2C09	2BHA	B		
585	S2-SI-149-H-011	2C09	2BHA	B		
586	S2-SI-149-H-013	2C09	2BHA	B		
587	S2-SI-149-H-015	2C09	2BHA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
588	S2-SI-149-H-018	2C09	2BHA	B		
589	S2-SI-164-H-005	2C09	2BHA	C		
590	S2-SI-065-H-028	2C06	2BHA	C		
591	S2-SI-077-H-019	2C06	2BHA	C		
592	S2-FS-107-H-002	2C11	2BHB	C		
593	S2-SI-107-H-006	2C11	2BHB	C		
594	S2-SI-107-H-007	2C07	2BHB	C		
595	S2-SI-080-H-011	2C03	2BHB	C		
596	S2-SI-080-H-013	2C02	2BHB	C		
597	S2-SI-080-H-028	2C02	2BHB	B		
598	S2-SI-080-H-030	2C02	2BHB	B		
599	S2-SI-080-H-033	2C02	2BHB	B		
600	S2-SI-080-H-047	2C07	2BHB	C		
601	S2-SI-080-H-050	2C07	2BHB	C		
602	S2-SI-131-H-006	2C02	2BHB	B		
603	S2-SI-131-H-009	2C02	2BHB	B		
604	S2-SI-151-H-007	2C02	2BHB	B		
605	S2-SI-151-H-008	2C02	2BHB	B		
606	S2-SI-151-H-011	2C02	2BHB	B		
607	S2-SI-151-H-013	2C02	2BHB	B		
608	S2-CS-001-H-015	2C01	2BKA	B		
609	S2-CS-011-H-021	2C02	2BKA	B		
610	S2-CS-011-H-022	2C02	2BKA	B		
611	S2-CS-002-H-014	2C02	2BKA	B		
612	S2-CS-003-H-011	2C01	2BKA	B		
613	S2-CS-003-H-041	2C05	2BKA	B		
614	S2-CS-003-H-043	2C05	2BKA	B		
615	S2-CS-003-H-044	2C05	2BKA	B		
616	S2-CS-003-H-046	2C05	2BKA	B		
617	S2-CS-003-H-047	2C05	2BKA	B		
618	S2-CS-003-H-053	2C01	2BKA	B		
619	S2-CS-003-H-059	2C02	2BKA	B		
620	S2-CS-003-H-061	2C01	2BKA	B		
621	S2-CS-003-H-063	2C01	2BKA	B		
622	S2-CS-003-H-065	2C01	2BKA	B		
623	S2-CS-003-H-068	2C02	2BKA	B		
624	S2-CS-003-H-075	2C02	2BKA	B		
625	S2-CS-003-H-077	2C02	2BKA	B		
626	S2-CS-004-H-006	2C02	2BKA	B		
627	S2-CS-004-H-014	2C05	2BKA	B		
628	S2-CS-004-H-028	2C09	2BKA	B		
629	S2-CS-004-H-034	2C01	2BKA	B		
630	S2-CS-004-H-039	2C01	2BKA	B		
631	S2-CS-004-H-043	2C01	2BKA	B		
632	S2-CS-004-H-048	2C09	2BKA	B		
633	S2-CS-004-H-053	2C15	2BKA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
634	S2-CS-004-H-054	2C13	2BKA	B		
635	S2-CS-004-H-056	2C01	2BKA	B		
636	S2-CS-004-H-058	2C02	2BKA	B		
637	S2-CS-004-H-062	2C02	2BKA	B		
638	S2-CS-004-H-074	2C02	2BKA	B		
639	S2-CS-004-H-079	2C02	2BKA	B		
640	S2-CS-004-H-080	2C01	2BKA	B		
641	S2-CS-018-H-002	2C02	2BKA	B		
642	S2-CS-027-H-014	2C02	2BKA	B		
643	S2-CS-027-H-015	2C02	2BKA	B		
644	S2-CS-027-H-020	2C02	2BKA	B		
645	S2-CS-027-H-022	2C02	2BKA	B		
646	S2-CS-028-H-007	2C02	2BKA	B		
647	S2-CS-032-H-005	2C02	2BKA	B		
648	S2-CS-032-H-006	2C02	2BKA	B		
649	S2-CS-032-H-008	2C02	2BKA	B		
650	S2-CS-033-H-004	2C02	2BKA	B		
651	S2-CS-041-H-023	2C09	2BKA	C		
652	S2-CS-047-H-017	2C05	2BKA	C		
653	S2-CS-047-H-022	2C05	2BKA	C		
654	S2-SI-004-H-032	2C02	2BKA	B		
655	S2-SI-004-H-034	2C02	2BKA	B		
656	S2-SI-114-H-001	2C02	2BKA	B		
657	S2-SI-115-H-117	2C02	2BKA	B		
658	S2-SI-115-H-020	2C02	2BKA	B		
659	S2-SI-117-H-001	2C02	2BKA	B		
660	S2-SI-117-H-002	2C02	2BKA	B		
661	S2-SI-119-H-002	2C02	2BKA	B		
662	S2-SI-119-H-003	2C02	2BKA	B		
663	S2-SI-119-H-004	2C02	2BKA	B		
664	S2-SI-119-H-007	2C02	2BKA	B		
665	S2-SI-119-H-008	2C92	2BKA	B		
666	S2-SI-114-H-003	2C02	2BKA	B		
667	S2-SI-115-H-025	2C02	2BKA	B		
668	S2-ST-015-H-004	2C05	2BMD	A		
669	S2-ST-015-H-005	2C06	2BMD	A		
670	S2-ST-015-H-012	2C06	2BMD	C		
671	S2-ST-015-H-013	2C06	2BMD	C		
672	S2-ST-015-H-014	2C06	2BMD	C		
673	S2-ST-106-H-009	2C09	2BMD	C		
674	S2-ST-106-H-010	2C05	2BMD	C		
675	S2-ST-106-H-011	2C05	2BMD	C		
676	S2-ST-106-H-012	2C05	2BMD	C		
677	S2-ST-106-H-013	2C05	2BMD	C		
678	S2-ST-106-H-014	2C05	2BMD	C		
679	S2-FS-007-H-003	2C09	2ECA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
680	S2-FS-007-H-005	2C09	2ECA	B		
681	S2-FS-007-H-007	2C09	2ECA	B		
682	S2-FS-007-H-017	2C05	2ECA	B		
683	S2-FS-007-H-018	2C05	2ECA	B		
684	S2-FS-007-H-022	2C09	2ECA	B		
685	S2-FS-025-H-005	2C13	2ECA	C		
686	S2-FS-030-H-002	2C13	2ECA	C		
687	S2-FS-139-H-001	2C09	2ECA	B		
688	S2-FS-139-H-002	2C09	2ECA	B		
689	S2-FS-139-H-008	2C09	2ECA	B		
690	S2-FS-139-H-013	2C13	2ECA	B		
691	S2-SI-133-H-020	2C01	2ECA	B		
692	S2-SI-133-H-024	2C01	2ECA	B		
693	S2-SI-133-H-028	2C05	2ECA	B		
694	S2-SI-133-H-030	2C05	2ECA	B		
695	S2-SI-133-H-039	2C05	2ECA	B		
696	S2-SI-133-H-047	2C09	2ECA	B		
697	S2-SI-133-H-049	2C13	2ECA	B		
698	S2-FS-007-H-009	2C09	2ECA	B		
699	SA-CC-112-H-002	CA06	2EGA	C		
700	SA-CC-127-H-009	CA05	2EGA	C		
701	S2-CC-004-H-006	2C06	2EGA	A		
702	S2-CC-004-H-007	2C06	2EGA	A		
703	S2-CC-005-H-008	2C01	2EGA	C		
704	S2-CC-006-H-009	2C09	2EGA	A		
705	S2-CC-007-H-008	2C02	2EGA	B		
706	S2-CC-008-H-008	2C09	2EGA	A		
707	S2-CC-008-H-010	2C09	2EGA	A		
708	S2-CC-008-H-018	2C09	2EGA	A		
709	S2-CC-008-H-019	2C09	2EGA	A		
710	S2-CC-008-H-020	2C05	2EGA	A		
711	S2-CC-008-H-021	2C09	2EGA	A		
712	S2-CC-010-H-001	2C10	2EGA	A		
713	S2-CC-013-H-017	2C10	2EGA	A		
714	S2-CC-014-H-002	2C05	2EGA	A		
715	S2-CC-014-H-003	2C05	2EGA	A		
716	S2-CC-014-H-005	2C05	2EGA	A		
717	S2-CC-014-H-006	2C05	2EGA	A		
718	S2-CC-146-H-005	2C01	2EGA	C		
719	S2-CC-149-H-002	2C01	2EGA	C		
720	S2-CC-266-H-025	2C13	2EGA	C		
721	S2-SC-001-H-001	2C01	2EPA	C		
722	S2-SC-001-H-016	2S01	2EPA	C		
723	S2-SC-002-H-005	2S01	2EPA	C		
724	S2-SC-002-H-008	2S02	2EPA	C		
725	S2-SC-003-H-011	2S01	2EPA	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
726	S2-ST-016-H-015	2C05	2BMD	C		
727	S2-ST-016-H-016	2C05	2BMD	C		
728	S2-CC-292-H-020	2C13	2EGA	C		
729	SA-CH-390-H-005	CA10	2GKA	C		
730	S2-CC-275-H-008	2C10	2GNF	C		
731	S2-CC-275-H-009	2C10	2GNF	C		
732	S2-CC-287-H-009	2C09	2GNF	C		
733	S2-CC-287-H-011	2C09	2GNF	C		
734	S2-CB-005-H-001	2C09	2GNF	C		
735	S2-CB-006-H-001	2C05	2GNF	C		
736	S2-GR-052-H-004	2C09	2HAA	C		
737	S2-GR-065-H-010	2C09	2HAA	C		
738	S2-GR-066-H-022	CA05	2HAA	A		
739	S3-GR-066-H-012	CA06	2HAA	A		
740	CEDM-1	Snubbers 740		A	I	YES
741	CEDM-2	through 847		A	I	YES
742	CEDM-3	are located on		A	I	YES
743	CEDM-4	CEDMs Below		A	I	YES
744	CEDM-5	Closure Head		A	I	YES
745	CEDM-6	Lift Rig		A	I	YES
746	CEDM-7	Seismic Plate		A	I	YES
747	CEDM-8			A	I	YES
748	CEDM-9			A	I	YES
749	CEDM-10			A	I	YES
750	CEDM-11			A	I	YES
751	CEDM-12			A	I	YES
752	CEDM-13			A	I	YES
753	CEDM-14			A	I	YES
754	CEDM-15			A	I	YES
755	CEDM-16			A	I	YES
756	CEDM-17			A	I	YES
757	CEDM-18			A	I	YES
758	CEDM-19			A	I	YES
759	CEDM-20			A	I	YES
760	CEDM-21			A	I	YES
761	CEDM-22			A	I	YES
762	CEDM-23			A	I	YES
763	CEDM-24			A	I	YES
764	CEDM-25			A	I	YES
765	CEDM-26			A	I	YES
766	CEDM-27			A	I	YES
767	CEDM-28			A	I	YES
768	CEDM-29			A	I	YES
769	CEDM-30			A	I	YES
770	CEDM-31			A	I	YES
771	CEDM-32			A	I	YES

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SAFETY RELATED MECHANICAL SNUBBERS*

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SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
772	CEDM-33			A	I	YES
773	CEDM-34			A	I	YES
774	CEDM-35			A	I	YES
775	CEDM-36			A	I	YES
776	CEDM-37			A	I	YES
777	CEDM-38			A	I	YES
778	CEDM-39			A	I	YES
779	CEDM-40			A	I	YES
780	CEDM-41			A	I	YES
781	CEDM-42			A	I	YES
782	CEDM-43			A	I	YES
783	CEDM-44			A	I	YES
784	CEDM-45			A	I	YES
785	CEDM-46			A	I	YES
786	CEDM-47			A	I	YES
787	CEDM-48			A	I	YES
788	CEDM-49			A	I	YES
789	CEDM-50			A	I	YES
790	CEDM-51			A	I	YES
791	CEDM-52			A	I	YES
792	CEDM-53			A	I	YES
793	CEDM-54			A	I	YES
794	CEDM-55			A	I	YES
795	CEDM-56			A	I	YES
796	CEDM-57			A	I	YES
797	CEDM-58			A	I	YES
798	CEDM-59			A	I	YES
799	CEDM-60			A	I	YES
800	CEDM-61			A	I	YES
801	CEDM-62			A	I	YES
802	CEDM-63			A	I	YES
803	CEDM-64			A	I	YES
804	CEDM-65			A	I	YES
805	CEDM-66			A	I	YES
806	CEDM-67			A	I	YES
807	CEDM-68			A	I	YES
808	CEDM-69			A	I	YES
809	CEDM-70			A	I	YES
810	CEDM-71			A	I	YES
811	CEDM-72			A	I	YES
812	CEDM-73			A	I	YES
813	CEDM-74			A	I	YES
814	CEDM-75			A	I	YES
815	CEDM-76			A	I	YES
816	CEDM-77			A	I	YES
817	CEDM-78			A	I	YES

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
818	CEDM-79			A	I	YES
819	CEDM-80			A	I	YES
820	CEDM-81			A	I	YES
821	CEDM-82			A	I	YES
822	CEDM-83			A	I	YES
823	CEDM-84			A	I	YES
824	CEDM-85			A	I	YES
825	CEDM-86			A	I	YES
826	CEDM-87			A	I	YES
827	CEDM-88			A	I	YES
828	CEDM-89			A	I	YES
829	CEDM-90			A	I	YES
830	CEDM-91			A	I	YES
831	CEDM-92			A	I	YES
832	CEDM-93			A	I	YES
833	CEDM-94			A	I	YES
834	CEDM-95			A	I	YES
835	CEDM-96			A	I	YES
836	CEDM-97			A	I	YES
837	CEDM-98			A	I	YES
838	CEDM-99			A	I	YES
839	CEDM-100			A	I	YES
840	CEDM-101			A	I	YES
841	CEDM-102			A	I	YES
842	CEDM-103			A	I	YES
843	CEDM-104			A	I	YES
844	CEDM-105			A	I	YES
845	CEDM-106			A	I	YES
846	CEDM-107			A	I	YES
847	CEDM-108			A	I	YES
848	CEDM-109	Snubbers 848		A	I	YES
849	CEDM-110	through 955		A	I	YES
850	CEDM-111	are located on		A	I	YES
851	CEDM-112	CEDMs Above		A	I	YES
852	CEDM-113	Closure Head		A	I	YES
853	CEDM-114	Lift Rig		A	I	YES
854	CEDM-115	Seismic Plate		A	I	YES
855	CEDM-116			A	I	YES
856	CEDM-117			A	I	YES
857	CEDM-118			A	I	YES
858	CEDM-119			A	I	YES
859	CEDM-120			A	I	YES
860	CEDM-121			A	I	YES
861	CEDM-122			A	I	YES
862	CEDM-123			A	I	YES
863	CEDM-124			A	I	YES

TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

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SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
864	CEDM-125			A	I	YES
865	CEDM-126			A	I	YES
866	CEDM-127			A	I	YES
867	CEDM-128			A	I	YES
868	CEDM-129			A	I	YES
869	CEDM-130			A	I	YES
870	CEDM-131			A	I	YES
871	CEDM-132			A	I	YES
872	CEDM-133			A	I	YES
873	CEDM-134			A	I	YES
874	CEDM-135			A	I	YES
875	CEDM-136			A	I	YES
876	CEDM-137			A	I	YES
877	CEDM-138			A	I	YES
878	CEDM-139			A	I	YES
879	CEDM-140			A	I	YES
880	CEDM-141			A	I	YES
881	CEDM-142			A	I	YES
882	CEDM-143			A	I	YES
883	CEDM-144			A	I	YES
884	CEDM-145			A	I	YES
885	CEDM-146			A	I	YES
886	CEDM-147			A	I	YES
887	CEDM-148			A	I	YES
888	CEDM-149			A	I	YES
889	CEDM-150			A	I	YES
890	CEDM-151			A	I	YES
891	CEDM-152			A	I	YES
892	CEDM-153			A	I	YES
893	CEDM-154			A	I	YES
894	CEDM-155			A	I	YES
895	CEDM-156			A	I	YES
896	CEDM-157			A	I	YES
897	CEDM-158			A	I	YES
898	CEDM-159			A	I	YES
899	CEDM-160			A	I	YES
900	CEDM-161			A	I	YES
901	CEDM-162			A	I	YES
902	CEDM-163			A	I	YES
903	CEDM-164			A	I	YES
904	CEDM-165			A	I	YES
905	CEDM-166			A	I	YES
906	CEDM-167			A	I	YES
907	CEDM-168			A	I	YES
908	CEDM-169			A	I	YES
909	CEDM-170			A	I	YES

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
910	CEDM-171			A	I	YES
911	CEDM-172			A	I	YES
912	CEDM-173			A	I	YES
913	CEDM-174			A	I	YES
914	CEDM-175			A	I	YES
915	CEDM-176			A	I	YES
916	CEDM-177			A	I	YES
917	CEDM-178			A	I	YES
918	CEDM-179			A	I	YES
919	CEDM-180			A	I	YES
920	CEDM-181			A	I	YES
921	CEDM-182			A	I	YES
922	CEDM-183			A	I	YES
923	CEDM-184			A	I	YES
924	CEDM-185			A	I	YES
925	CEDM-186			A	I	YES
926	CEDM-187			A	I	YES
927	CEDM-188			A	I	YES
928	CEDM-189			A	I	YES
929	CEDM-190			A	I	YES
930	CEDM-191			A	I	YES
931	CEDM-192			A	I	YES
932	CEDM-193			A	I	YES
933	CEDM-194			A	I	YES
934	CEDM-195			A	I	YES
935	CEDM-196			A	I	YES
936	CEDM-197			A	I	YES
937	CEDM-198			A	I	YES
938	CEDM-199			A	I	YES
939	CEDM-200			A	I	YES
940	CEDM-201			A	I	YES
941	CEDM-202			A	I	YES
942	CEDM-203			A	I	YES
943	CEDM-204			A	I	YES
944	CEDM-205			A	I	YES
945	CEDM-206			A	I	YES
946	CEDM-207			A	I	YES
947	CEDM-208			A	I	YES
948	CEDM-209			A	I	YES
949	CEDM-210			A	I	YES
950	CEDM-211			A	I	YES
951	CEDM-212			A	I	YES
952	CEDM-213			A	I	YES
953	CEDM-214			A	I	YES
954	CEDM-215			A	I	YES
955	CEDM-216			A	I	YES

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

Radiation Zone Key for Table 3.7-4b

A - High radiation zone (>100 MR/HR)

B - Low radiation zone during normal operation. Higher radiation zone during shutdown.

C - Low radiation zone (<100 MR/HR)

D - Consider plant operation and/or special draining procedures for access.

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
1	S2-RC-042-H-00C	2C06	2BBA	A		
2	S2-RC-140-H-00M	2C09	2BBA	A		
3	S2-RC-149-H-00A	2C09	2BBA	A		
4	S2-RC-149-H-00C	2C09	2BBA	A		
5	S2-RC-150-H-00B	2C10	2BBA	A		
6	S2-RC-157-H-00L	2C09	2BBA	A		
7	S2-RC-017-H-00A	2C05	2BHA	B		
8	S2-RC-017-H-00B	2C05	2BHA	B		
9	S3-ST-016-H-00C	2C05	2BMD	C		
10	S3-ST-001-H-00F	3C06	3ABB	C		
11	S3-BM-028-H-0AM	CA03	3BGB	D		
12	S3-CC-016-H-00R	3C01	3EGA	B		
13	S3-CC-016-H-00Y	3C01	3EGA	B		
14	S3-SS-030-H-00M	3C01	3EGA	B		
15	S2-SS-030-H-003	2C05	2SJA	A		
16	S2-SS-030-H-005	2C09	2SJA	A		
17	S2-SS-030-H-007	2C09	2SJA	A		
18	S2-SS-030-H-013	2C09	2SJA	A		
19	S2-SS-030-H-021	2C09	2SJA	A		
20	S2-SS-030-H-035	2C09	2SJA	A		
21	S2-SS-030-H-039	2C09	2SJA	A		
22	S2-SS-030-H-045	2C09	2SJA	A		

*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.

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
23	S2-SS-030-H-008	2C09	2SJA	B		
24	S2-SS-030-H-006	2C09	2SJA	C		
25	S2-SS-030-H-008	2C09	2SJA	C		
26	S2-SS-030-h-016	2C09	2SJA	A		
27	S2-SS-030-H-019	2C09	2SJA	C		
28	S2-SS-030-H-023	2C09	2SJA	C		
29	S2-SS-030-H-030	2C09	2SJA	C		
30	S2-SS-037-H-010	2C09	2SJA	C		
31	S2-SS-037-H-010	2C09	2SJA	C		
32	S2-SS-037-H-011	2C09	2SJA	C		
33	S2-SS-042-H-004	2C09	2SJA	C		
34	S2-SS-042-H-010	2C09	2SJA	C		
35	S2-SS-042-H-018	2C09	2SJA	C		
36	S2-SS-042-H-021	2C09	2SJA	A		
37	S2-SS-043-H-003	2C09	2SJA	C		
38	S2-SS-167-H-001	2C10	2SJA	A		
39	S2-SS-167-H-003	2C10	2SJA	A		
40	S2-SS-167-H-005	2C06	2SJA	A		
41	S2-SS-167-H-009	2C06	2SJA	A		
42	S2-SS-167-H-022	2C05	2SJA	A		
43	S2-SS-167-H-043	2C09	2SJA	A		
44	S2-SS-167-H-047	2C09	2SJA	C		
45	S2-SS-167-H-062	2C05	2SJA	A		
46	S2-SS-167-H-064	2C05	2SJA	A		
47	S2-SS-167-H-069	2C05	2SJA	A		
48	S2-SS-167-H-074	2C05	2SJA	A		
49	S2-SS-168-H-008	2C09	2SJA	C		
50	S2-SS-037-H-006	2C09	2SJA	B		
51	S2-SS-043-H-004	2C09	2SJA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
52	S2-SI-045-H-028	2C05	2SJA	C		
53	S2-ST-001-H-014	2C02	SABA	C		
54	S2-ST-001-H-021	2C02	2ABA	C		
55	S2-ST-001-H-022	2C02	2ABA	C		
56	S2-ST-001-H-023	2C02	2ABA	C		
57	S2-ST-001-H-031	2C02	2ABA	C		
58	S2-ST-001-H-034	2C02	2ABA	C		
59	S2-ST-002-H-014	2C01	2ABA	C		
60	S2-ST-002-H-026	2C01	2ABA	C		
61	S2-ST-002-H-034	2C01	2ABA	C		
62	S2-ST-017-H-004	2C06	2ABA	C		
63	S2-ST-017-H-005	2C06	2ABA	C		
64	S2-ST-017-H-008	2C06	2ABA	C		
65	S2-ST-017-H-009	2C06	2ABA	C		
66	S2-ST-577-H-014	2C01	2ABA	C		
67	S2-ST-577-H-018	2C02	2ABA	C		
68	S2-ST-577-H-021	2C02	2ABA	C		
69	S2-ST-577-H-032	2C02	2ABA	C		
70	S2-FW-189-H-010	2C06	2ABB	C		
71	S2-FW-189-H-012	2C06	2ABB	C		
72	S2-FW-189-H-013	2C06	2ABB	C		
73	S2-FW-189-H-014	2C06	2ABB	C		
74	S2-FW-189-H-017	2C06	2ABB	C		
75	S2-FW-190-H-011	2C05	2ABB	C		
76	S2-FW-190-H-013	2C05	2ABB	C		
77	S2-FW-190-H-014	2C05	2ABB	C		
78	S2-FW-190-H-015	2C05	2ABB	C		
79	S2-FW-190-H-018	2C05	2ABB	C		
80	S2-ST-001-H-001	2C06	2ABB	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
81	S2-ST-001-H-002	2C06	2ABB	C		
82	S2-ST-001-H-004	2C06	2ABB	C		
83	S2-ST-001-H-005	2C06	2ABB	C		
84	S2-ST-001-H-007	2C06	2ABB	C		
85	S2-ST-001-H-008	2C06	2ABB	C		
86	S2-ST-001-H-009	2C06	2ABB	C		
87	S2-ST-001-H-025	2C06	2ABB	C		
88	S2-ST-001-H-040	2C02	2ABB	C		
89	S2-ST-001-H-042	2C06	2ABB	C		
90	S2-ST-001-H-044	2C02	2ABB	C		
91	S2-ST-001-H-048	2C02	2ABB	C		
92	S2-ST-001-H-049	2C02	2ABB	C		
93	S2-ST-002-H-001	2C05	2ABB	C		
94	S2-ST-002-H-002	2C05	2ABB	C		
95	S2-ST-002-H-004	2C05	2ABB	C		
96	S2-ST-002-H-005	2C05	2ABB	C		
97	S2-ST-002-H-007	2C05	2ABB	C		
98	S2-ST-002-H-008	2C05	2ABB	C		
99	S2-ST-002-H-009	2C05	2ABB	C		
100	S2-ST-002-H-022	2C05	2ABB	C		
101	S2-ST-002-H-040	2C01	2ABB	C		
102	S2-ST-002-H-041	2C01	2ABB	C		
103	S2-ST-002-H-042	2C01	2ABB	C		
104	S2-ST-002-H-046	2C01	2ABB	C		
105	S2-ST-022-H-047	2C01	2ABB	C		
106	S2-ST-004-H-001	2C01	2ABB	C		
107	S2-ST-004-H-003	2C01	2ABB	C		
108	S2-ST-014-H-011	2C10	2ABB	C		
109	S2-ST-014-H-014	2C10	2ABB	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
110	S2-ST-014-H-022	2C06	2ABB	A		
111	S2-ST-014-H-023	2C06	2ABB	A		
112	S2-ST-014-H-039	2C10	2ABB	C		
113	S2-ST-014-H-041	2C06	2ABB	C		
114	S2-ST-014-H-042	2C06	2ABB	C		
115	S2-ST-014-H-051	2C10	2ABB	C		
116	S2-ST-014-H-052	2C10	2ABB	C		
117	S2-ST-015-H-016	2C06	2ABB	A		
118	S2-ST-015-H-017	2C06	2ABB	A		
119	S2-ST-015-H-018	2C06	2ABB	A		
120	S2-ST-015-H-019	2C06	2ABB	A		
121	S2-ST-016-H-002	2C05	2ABB	A		
122	S2-ST-016-H-003	2C09	2ABB	A		
123	S2-ST-016-H-031	2C09	2ABB	A		
124	S2-ST-018-H-002	2C09	2ABB	C		
125	S2-ST-018-H-006	2C09	2ABB	C		
126	S2-ST-018-H-007	2C09	2ABB	C		
127	S2-ST-018-H-009	2C09	2ABB	C		
128	S2-ST-018-H-020	2C09	2ABB	A		
129	S2-ST-018-H-033	2C09	2ABB	A		
130	S2-ST-018-H-035	2C09	2ABB	A		
131	S2-ST-018-H-036	2C09	2ABB	A		
132	S2-ST-018-H-039	2C09	2ABB	A		
133	S2-ST-309-H-003	2C01	2ABB	C		
134	S2-ST-578-H-001	2C02	2ABB	C		
135	S2-ST-578-H-003	2C02	2ABB	C		
136	S2-ST-580-H-002	2C02	2ABB	C		
137	S2-ST-583-H-003	2C01	2ABB	C		
138	S2-FW-222-H-004	2C06	2ALA	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
139	S2-FW-222-H-008	2C10	2ALA	C		
140	S2-FW-222-H-010	2C06	2ALA	C		
141	S2-FW-223-H-002	2C05	2ALA	C		
142	S2-F2-223-H-003	2C05	2ALA	C		
143	S2-FW-223-H-007	2C05	2ALA	C		
144	S2-FW-223-H-012	2C05	2ALA	C		
145	S2-FW-301-H-003	2C12	2ANA	C		
146	S2-FW-301-H-005	2C12	2ANA	C		
147	S2-FW-301-H-008	2C12	2ANA	C		
148	S2-FW-302-H-003	2C12	2ANA	C		
149	S2-CC-070-H-004	2C05	2BBA	A		
150	S2-CC-072-H-005	2C05	2BBA	A		
151	S2-CC-072-H-006	2C05	2BBA	A		
152	S2-CC-074-H-004	2C05	2BBA	A		
153	S2-CC-074-H-008	2C05	2BBA	A		
154	S2-CC-074-H-010	2C09	2BBA	A		
155	S2-CC-076-H-006	2C05	2BBA	A		
156	S2-CC-076-H-009	2C05	2BBA	A		
157	S2-CC-076-H-012	2C09	2BBA	A		
158	S2-CC-078-H-006	2C06	2BBA	A		
159	S2-CC-078-H-008	2C10	2BBA	A		
160	S2-CC-078-H-010	2C10	2BBA	A		
161	S2-CC-080-H-007	2C06	2BBA	A		
162	S2-CC-080-H-008	2C10	2BBA	A		
163	S2-CC-080-H-010	2C10	2BBA	A		
164	S2-CC-082-H-007	2C06	2BBA	A		
165	S2-CC-082-H-008	2C06	2BBA	A		
166	S2-CC-082-H-009	2C06	2BBA	A		
167	S2-CC-082-H-011	2C10	2BBA	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
168	S2-CC-082-H-012	2C06	2BBA	A		
169	S2-CC-082-H-013	2C06	2BBA	A		
170	S2-CC-084-H-004	2C06	2BBA	A		
171	S2-CC-084-H-005	2C06	2BBA	A		
172	S2-CC-084-H-006	2C06	2BBA	A		
173	S2-CC-084-H-008	2C06	2BBA	A		
174	S2-CC-084-H-009	2C06	2BBA	A		
175	S2-CC-084-H-010	2C06	2BBA	A		
176	S2-CC-364-H-002	2C09	2BBA	A		
177	S2-CC-365-H-003	2C09	2BBA	A		
178	S2-CC-366-H-003	2C05	2BBA	A		
179	S2-CC-367-H-002	2C06	2BBA	A		
180	S2-CC-368-H-001	2009	2BBA	A		
181	S2-CC-371-H-001	2C06	2BBA	A		
182	S2-CC-372-H-001	2C09	2BBA	A		
183	S2-CC-372-H-003	2C09	2BBA	A		
184	S2-CC-373-H-002	2C10	2BBA	A		
185	S2-CC-374-H-001	2C05	2BBA	A		
186	S2-CC-375-H-002	2C06	2BBA	A		
187	S2-CC-375-H-003	2C06	2BBA	A		
188	S2-RC-018-H-001	2C05	2BBA	A		
189	S2-RC-021-H-001	2C05	2BBA	A		
190	S2-RC-021-H-002	2C05	2BBA	A		
191	S2-RC-022-H-002	2C05	2BBA	A		
192	S2-RC-022-H-004	2C05	2BBA	A		
193	S2-RC-022-H-010	2C05	2BBA	A		
194	S2-RC-031-H-001	2C10	2BBA	A		
195	S2-RC-052-H-002	2C09	2BBA	B		
196	S2-RC-011-H-001	2C09	2BBA	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
197	S2-RC-012-H-026	2C09	2BBA	C		
198	S2-RC-012-H-029	2C09	2BBA	C		
199	S2-RC-013-H-002	2C05	2BBA	A		
200	S2-RC-015-H-017	2C05	2BBA	A		
201	S2-RC-015-H-018	2C05	2BBA	A		
202	S2-RC-015-H-020	2C09	2BBA	A		
203	S2-RC-016-H-004	2C06	2BBA	A		
204	S2-RC-016-H-008	2C05	2BBA	A		
205	S2-RC-016-H-009	2C06	2BBA	A		
206	S2-RC-016-H-010	2C06	2BBA	A		
207	S2-RC-023-H-001	2C05	2BBB	A		
208	S2-RC-032-H-002	2C09	2BBB	C		
209	S2-RC-033-H-022	2C09	2BBB	C		
210	S2-RC-034-H-012	2C09	2BBB	C		
211	S2-RC-034-H-014	2C09	2BBB	C		
212	S2-RC-034-H-015	2C09	2BBB	C		
213	S2-RC-034-H-018	2C09	2BBB	C		
214	S2-RC-034-H-021	2C09	2BBB	C		
215	S2-RC-034-H-023	2C09	2BBB	C		
216	S2-RC-049-H-002	2C09	2BBB	A		
217	S2-RC-060-H-003	2C09	2BBB	A		
218	S2-RC-060-H-009	2C09	2BBB	A		
219	S2-RC-060-H-012	2C09	2BBB	A		
220	S2-RC-061-H-012	2C10	2BBB	A		
221	S2-RC-061-H-013	2C06	2BBB	A		
222	S2-RC-061-H-014	2C06	2BBB	A		
223	S2-RC-061-H-016	2C06	2BBB	A		
224	S2-RC-062-H-003	2C09	2BBB	A		
225	S2-RC-062-H-009	2C09	2BBB	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
226	S2-RC-062-H-012	2C09	2BBB	A		
227	S2-RC-062-H-013	2C09	2BBB	A		
228	S2-RC-062-H-014	2C09	2BBB	C		
229	S2-RC-062-H-015	2C09	2BBB	C		
230	S2-RC-064-H-002	2C06	2BBB	A		
231	S2-RC-064-H-003	2C06	2BBB	A		
232	S2-RC-072-H-002	2C06	2BBB	C		
233	S2-RC-073-H-019	2C05	2BBB	A		
234	S2-RC-073-H-021	2C05	2BBB	A		
235	S2-RC-073-H-023	2C05	2BBB	A		
236	S2-RC-073-H-025	2C09	2BBB	A		
237	S2-RC-073-H-026	2C06	2BBB	A		
238	S2-RC-073-H-031	2C06	2BBB	C		
239	S2-RC-094-H-009	2C09	2BBB	C		
240	S2-RC-096-H-025	2C09	2BBB	C		
241	S2-RC-147-H-001	2C05	2BBB	A		
242	S2-SI-021-H-026	2C14	2BBB	B		
243	S2-SI-043-H-016	2C09	2BBB	C		
244	S2-SI-043-H-018	2C09	2BBB	A		
245	S2-SI-043-H-020	2C09	2BBB	A		
246	S2-SI-043-H-021	2C09	2BBB	A		
247	S2-SI-044-H-001	2C05	2BBB	C		
248	S2-SI-044-H-003	2C05	2BBB	C		
249	S2-SI-044-H-004	2C05	2BBB	C		
250	S2-SI-044-H-006	2C05	2BBB	A		
251	S2-SI-044-H-009	2C05	2BBB	A		
252	S2-SI-045-H-005	2C05	2BBB	C		
253	S2-SI-045-H-008	2C05	2BBB	C		
254	S2-SI-045-H-011	2C06	2BBB	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
226	S2-RC-062-H-012	2C09	2BBB	A		
227	S2-RC-062-H-013	2C09	2BBB	A		
228	S2-RC-062-H-014	2C09	2BBB	C		
229	S2-RC-062-H-015	2C09	2BBB	C		
230	S2-RC-064-H-002	2C06	2BBB	A		
231	S2-RC-064-H-003	2C06	2BBB	A		
232	S2-RC-072-H-002	2C06	2BBB	C		
233	S2-RC-073-H-019	2C05	2BBB	A		
234	S2-RC-073-H-021	2C05	2BBB	A		
235	S2-RC-073-H-023	2C05	2BBB	A		
236	S2-RC-073-H-025	2C09	2BBB	A		
237	S2-RC-073-H-026	2C06	2BBB	A		
238	S2-RC-073-H-031	2C06	2BBB	C		
239	S2-RC-094-H-009	2C09	2BBB	C		
240	S2-RC-096-H-025	2C09	2BBB	C		
241	S2-RC-147-H-001	2C05	2BBB	A		
242	S2-SI-021-H-026	2C14	2BBB	B		
243	S2-SI-043-H-016	2C09	2BBB	C		
244	S2-SI-043-H-018	2C09	2BBB	A		
245	S2-SI-043-H-020	2C09	2BBB	A		
246	S2-SI-043-H-021	2C09	2BBB	A		
247	S2-SI-044-H-001	2C05	2BBB	C		
248	S2-SI-044-H-003	2C05	2BBB	C		
249	S2-SI-044-H-004	2C05	2BBB	C		
250	S2-SI-044-H-006	2C05	2BBB	A		
251	S2-SI-044-H-009	2C05	2BBB	A		
252	S2-SI-045-H-005	2C05	2BBB	C		
253	S2-SI-045-H-008	2C05	2BBB	C		
254	S2-SI-045-H-011	2C06	2BBB	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
255	S2-SI-045-H-015	2C06	2BBB	A		
256	S2-SI-045-H-018	2C06	2BBB	A		
257	S2-SI-046-H-002	2C06	2BBB	A		
258	S2-SI-046-H-003	2C10	2BBB	C		
259	S2-SI-046-H-006	2C10	2BBB	C		
260	S2-SI-046-H-010	2C10	2BBB	A		
261	S2-SI-046-H-012	2C10	2BBB	A		
262	S2-SI-062-H-001	2C10	2BBB	A		
263	S2-SI-062-H-002	2C10	2BBB	A		
264	S2-SI-063-H-020	2C09	2BBB	A		
265	S2-SI-064-H-007	2C05	2BBB	A		
266	S2-SI-064-H-013	2C05	2BBB	A		
267	S2-SI-064-H-019	2C05	2BBB	C		
268	S2-SI-064-H-020	2C05	2BBB	A		
269	S2-SI-065-H-030	2C05	2BBB	C		
270	S2-SI-065-H-002	2C06	2BBB	A		
271	S2-SI-065-H-003	2C06	2BBB	A		
272	S2-SI-065-H-004	2C06	2BBB	A		
273	S2-SI-065-H-005	2C06	2BBB	A		
274	S2-SI-065-H-009	2C06	2BBB	A		
275	S2-SI-065-H-010	2C06	2BBB	A		
276	S2-SI-065-H-014	2C06	2BBB	A		
277	S2-SI-065-H-017	2C06	2BBB	A		
278	S2-SI-065-H-019	2C06	2BBB	A		
279	S2-SI-065-H-021	2C06	2BBB	A		
280	S2-SI-065-H-024	2C06	2BBB	A		
281	S2-SI-066-H-005	2C10	2BBB	C		
282	S2-SI-066-H-010	2C10	2BBB	A		
283	S2-SI-066-H-013	2C10	2BBB	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

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	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
284	S2-SI-066-H-015	2C10	2BBB	A		
285	S2-SI-066-H-018	2C10	2BBB	A		
286	S2-SI-066-H-022	2C10	2BBB	A		
287	S2-SI-066-H-024	2C10	2BBB	A		
288	S2-SI-066-H-025	2C10	2BBB	A		
289	S2-SI-066-H-026	2C10	2BBB	A		
290	S2-SI-066-H-028	2C10	2BBB	A		
291	S2-SI-066-H-030	2C10	2BBB	A		
292	S2-SI-087-H-007	2C05	2BBB	C		
293	S2-SI-087-H-011	2C05	2BBB	C		
294	S2-SI-087-H-019	2C06	2BBB	C		
295	S2-SI-087-H-021	2C06	2BBB	C		
296	S2-SI-087-H-022	2C06	2BBB	C		
297	S2-SI-087-H-025	2C06	2BBB	C		
298	S2-SI-087-H-026	2C06	2BBB	C		
299	S2-SI-152-H-001	2C09	2BBB	C		
300	S2-SI-155-H-001	2C10	2BBB	C		
301	S2-SI-165-H-006	2C09	2BBB	A		
302	S2-SI-165-H-007	2C09	2BBB	A		
303	S2-SI-166-H-002	2C09	2BBB	C		
304	S2-SI-168-H-001	2C06	2BBB	C		
305	S2-SI-168-H-002	2C06	2BBB	C		
306	S2-SI-168-H-003	2C06	2BBB	C		
307	S2-RC-034-H-010	2C09	2BBB	C		
308	S2-RC-061-H-010	2C06	2BBB	A		
309	S2-SI-045-H-013	2C06	2BBB	C		
310	S2-SI-063-H-022	2C09	2BBB	A		
311	S2-SI-063-H-023	2C09	2BBB	A		
312	S2-BM-021-H-003	CA03	2BGA	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
313	S2-BM-022-H-010	CA03	2BGA	D		
314	S2-BM-022-H-011	CA03	2BGA	D		
315	S2-BM-022-H-012	CA03	2BGA	D		
316	S2-BM-022-H-014	CA03	2BGA	D		
317	S2-BM-031-H-008	CA03	2BGA	D		
318	S2-BM-031-H-018	CA03	2BGA	D		
319	S2-BM-031-H-019	CA03	2BGA	D		
320	S2-BM-036-H-003	CA03	2BGA	D		
321	S2-BM-036-H-003	CA03	2BGA	D		
322	S2-BM-036-H-005	CA03	2GBA	D		
323	S2-BM-035-H-007	CA03	2BGA	C		
324	S2-FS-057-H-015	CA03	2BGA	D		
325	S2-FS-057-H-059	2C05	2BGA	C		
326	S2-FS-057-H-061	2C05	2BGA	C		
327	S2-FS-057-H-068	2C09	2BGA	C		
328	S2-FS-057-H-070	2C09	2BGA	C		
329	S2-FS-057-H-075	CA03	2BGA	C		
330	S2-FS-057-H-078	CA03	2BGA	C		
331	S2-FS-057-H-079	2C09	2BGA	C		
332	S2-FS-057-H-080	2C09	2BGA	C		
333	S2-FS-057-H-082	2C09	2BGA	C		
334	S2-FS-057-H-083	2C09	2BGA	C		
335	S2-FS-057-H-085	2C09	2BGA	C		
336	S2-FS-057-H-086	2C09	2BGA	D		
337	S2-FS-057-H-087	2C09	2BGA	C		
338	S2-FS-057-H-098	2C01	2BGA	C		
339	S2-RC-022-H-009	2C05	2BGA	A		
340	S2-VC-001-H-008	2C05	2BGA	A		
341	S2-VC-001-H-009	2C05	2BGA	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
342	S2-VC-001-H-010	2C05	2BGA	A		
343	S2-VC-001-H-013	2C05	2BGA	A		
344	S2-VC-001-H-014	2C05	2BGA	A		
345	S2-VC-001-H-015	2C05	2BGA	A		
346	S2-VC-001-H-016	2C05	2BGA	A		
347	S2-VC-003-H-019	2C09	2BGA	B		
348	S2-VC-004-H-017	CA05	2BGA	D		
349	S2-VC-007-H-005	CA03	2BGA	A		
350	S2-VC-007-H-017	CA03	2BGA	A		
351	S2-VC-007-H-018	CA03	2BGA	D		
352	S2-VC-007-H-025	CA03	2BGA	D		
353	S2-VC-008-H-039	2C09	2BGA	B		
354	S2-VC-009-H-010	2C09	2BGA	A		
355	S2-VC-009-H-011	2C09	2BGA	A		
356	S2-VC-009-H-012	2C09	2BGA	C		
357	S2-VC-010-H-006	2C05	2BGA	A		
358	S2-VC-010-H-014	2C09	2BGA	A		
359	S2-VC-010-H-020	2C06	2BGA	A		
360	S2-VC-010-H-022	2C06	2BGA	A		
361	S2-VC-010-H-027	2C06	2BGA	A		
362	S2-VC-010-H-029	2C06	2BGA	C		
363	S2-VC-010-H-031	2C05	2BGA	C		
364	S2-VC-011-H-011	2C09	2BGA	C		
365	S2-VC-014-H-002	CA03	2BGA	D		
366	S2-VC-035-H-006	CA03	2BGA	D		
367	S2-VC-046-H-001	2C09	2BGA	C		
368	S2-VC-055-H-015	CA03	2BGA	D		
369	S2-VC-055-H-017	CA03	2BGA	D		
370	S2-VC-055-H-020	CA03	2BGA	D		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
371	S2-VC-055-H-021	CA03	2BGA	D		
372	S2-VC-055-H-022	CA03	2BGA	D		
373	S2-VC-055-H-023	CA03	2BGA	D		
374	S2-VC-056-H-012	CA03	2BGA	D		
375	S2-VC-086-H-002	CA03	2BGA	D		
376	S2-VC-099-H-001	CA03	2BGA	D		
377	S2-VC-104-H-009	2C09	2BGA	C		
378	S2-VC-104-H-013	2C09	2BGA	A		
379	S2-VC-104-H-014	2C09	2BGA	A		
380	S2-VC-104-H-019	2C09	2BGA	B		
381	S2-RC-022-H-008	2C05	2BGA	A		
382	S2-VC-003-H-017	CA05	2BGA	A		
383	S2-VC-004-H-002	CA03	2BGA	A		
384	S2-VC-004-H-004	CA03	2BGA	A		
385	S2-VC-004-H-006	CA03	2BGA	A		
386	S2-VC-004-H-014	CA05	2BGA	A		
387	S2-VC-008-H-004	CA03	2BGA	D		
388	S2-VC-008-H-007	CA03	2BGA	D		
389	S2-VC-008-H-011	CA03	2BGA	D		
390	S2-VC-008-H-029	CA05	2BGA	A		
391	S2-VC-010-H-013	2C09	2BGA	A		
392	S2-VC-010-H-032	2C05	2BGA	A		
393	S2-VC-012-H-007	CA05	2BGA	A		
394	S2-VC-015-H-007	CA05	2BGA	D		
395	S2-VC-015-H-009	CA05	2BGA	A		
396	S2-VC-015-H-011	CA05	2BGA	A		
397	S2-VC-015-H-012	CA05	2BGA	A		
398	S2-VC-047-H-004	CA03	2BGA	D		
399	S2-VC-058-H-004	CA03	2BGA	D		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
400	S2-VC-059-H-005	CA03	2BGA	D		
401	S2-BM-003-H-008	CA03	2BGB	D		
402	S2-BM-004-H-007	CA03	2BGB	D		
403	S2-BM-004-H-009	CA03	2BGB	D		
404	S2-BM-024-H-006	CA03	2BGB	D		
405	S2-BM-025-H-008	CA03	2BGB	D		
406	S2-BM-037-H-004	CA03	2BGB	C		
407	S2-BM-037-H-011	CA03	2BGB	C		
408	S2-BM-037-H-017	CA03	2BGB	C		
409	S2-BM-037-H-019	CA03	2BGB	C		
410	S2-BM-037-H-020	CA03	2BGB	C		
411	S2-BM-037-H-022	CA03	2BGB	C		
412	S2-BM-056-H-002	CA03	2BGB	D		
413	S2-BM-057-H-002	CA03	2BGB	D		
414	S2-BM-057-H-003	CA03	2BGB	D		
415	S2-BM-057-H-005	CA03	2BGB	D		
416	S2-BM-057-H-006	CA03	2BGB	D		
417	S2-BM-057-H-009	CA05	2BGB	D		
418	S2-VC-116-H-007	CA03	2BGA	D		
419	S2-CS-019-H-009	2C01	2BHA	C		
420	S2-CS-020-H-001	2C02	2BHA	C		
421	S2-CS-020-H-003	2C02	2BHA	C		
422	S2-RC-016-H-003	2C06	2BHA	C		
423	S2-RC-017-H-002	2C06	2BHA	C		
424	S2-RC-017-H-030	2C06	2BHA	C		
425	S2-RC-017-H-033	2C06	2BHA	C		
426	S2-RC-017-H-036	2C05	2BHA	C		
427	S2-RC-017-H-038	2C05	2BHA	C		
428	S2-RC-017-H-040	2C05	2BHA	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
429	S2-RC-017-H-044	2C01	2BHA	B		
430	S2-RC-017-H-050	2C05	2BHA	B		
431	S2-RC-017-H-052	2C05	2BHA	B		
432	S2-RC-017-H-054	2C05	2BHA	B		
433	S2-RC-017-H-055	2C05	2BHA	B		
434	S2-RC-017-H-056	2C05	2BHA	B		
435	S2-RC-017-H-057	2C05	2BHA	B		
436	S2-RC-017-H-067	2C05	2BHA	B		
437	S2-RC-017-H-069	2C05	2BHA	B		
438	S2-RC-017-H-070	2C05	2BHA	B		
439	S2-RC-017-H-071	2C05	2BHA	B		
440	S2-RC-017-H-072	2C05	2BHA	B		
441	S2-RC-017-H-076	2C01	2BHA	B		
442	S2-RC-017-H-077	2C01	2BHA	B		
443	S2-RC-017-H-080	2C06	2BHA	C		
444	S2-RC-017-H-081	2C06	2BHA	C		
445	S2-RC-017-H-084	2C05	2BHA	B		
446	S2-RC-017-H-085	2C05	2BHA	B		
447	S2-RC-017-H-090	2C01	2BHA	B		
448	S2-RC-017-H-091	2C05	2BHA	B		
449	S2-RC-025-H-002	2C05	2BHA	B		
450	S2-RC-025-H-006	2C05	2BHA	B		
451	S2-RC-025-H-009	2C05	2BHA	B		
452	S2-RC-025-H-011	2C05	2BHA	B		
453	S2-RC-025-H-013	2C05	2BHA	B		
454	S2-RC-027-H-004	2C05	2BHA	B		
455	S2-RC-027-H-005	2C05	2BHA	B		
456	S2-RC-050-H-005	2C02	2BHA	B		
457	S2-RC-050-H-007	2C02	2BHA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
458	S2-RC-050-H-014	2C02	2BHA	B		
459	S2-RC-050-H-019	2C02	2BHA	B		
460	S2-RC-050-H-021	2C02	2BHA	B		
461	S2-RC-051-H-001	2C06	2BHA	C		
462	S2-RC-071-H-001	2C06	2BHA	C		
463	S2-RC-071-H-003	2C06	2BHA	C		
464	S2-RC-071-H-004	2C06	2BHA	C		
465	S2-RC-072-H-004	2C06	2BHA	C		
466	S2-RC-076-H-001	2C06	2BHA	C		
467	S2-RC-134-H-001	2C05	2BHA	B		
468	S2-RC-134-H-002	2C05	2BHA	B		
469	S2-RC-134-H-003	2C05	2BHA	B		
470	S2-RC-134-H-004	2C05	2BHA	B		
471	S2-RC-134-H-006	2C06	2BHA	B		
472	S2-RC-134-H-007	2C05	2BHA	B		
473	S2-SI-001-H-024	2C02	2BHA	B		
474	S2-SI-002-H-024	2C07	2BHA	C		
475	S2-SI-002-H-029	2C07	2BHA	C		
476	S2-SI-003-H-003	2C06	2BHA	C		
477	S2-SI-003-H-006	2C02	2BHA	C		
478	S2-SI-003-H-017	2C01	2BHA	B		
479	S2-SI-003-H-018	2C01	2BHA	B		
480	S2-SI-004-H-003	2C06	2BHA	C		
481	S2-SI-004-H-006	2C02	2BHA	C		
482	S2-SI-007-H-004	2C04	2BHA	C		
483	S2-SI-008-H-003	2C02	2BHA	B		
484	S2-SI-009-H-003	2C02	2BHA	B		
485	S2-SI-009-H-011	2C01	2BHA	B		
486	S2-SI-010-H-001	2C02	2BHA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
487	S2-SI-010-H-003	2C02	2BHA	B		
488	S2-SI-010-H-006	2C02	2BHA	B		
489	S2-SI-010-H-012	2C02	2BHA	B		
490	S2-SI-012-H-004	2C05	2BHA	B		
491	S2-SI-014-H-012	2C02	2BHA	B		
492	S2-SI-014-H-018	2C01	2BHA	B		
493	S2-SI-014-H-019	2C01	2BHA	B		
494	S2-SI-014-H-021	2C01	2BHA	B		
495	S2-SI-017-H-006	2C05	2BHA	B		
496	S2-SI-020-H-033	2C05	2BHA	C		
497	S2-SI-020-H-034	2C05	2BHA	C		
498	S2-SI-020-H-037	2C05	2BHA	C		
499	S2-SI-020-H-041	2C09	2BHA	C		
500	S2-SI-020-H-053	2C09	2BHA	C		
501	S2-SI-020-H-059	2C14	2BHA	B		
502	S2-SI-020-H-062	2C14	2BHA	B		
503	S2-SI-020-H-065	2C13	2BHA	C		
504	S2-SI-020-H-069	2C13	2BHA	C		
505	S2-SI-020-H-071	2C14	2BHA	B		
506	S2-SI-021-H-002	2C05	2BHA	B		
507	S2-SI-021-H-011	2C09	2BHA	B		
508	S2-SI-021-H-014	2C09	2BHA	B		
509	S2-SI-021-H-017	2C09	2BHA	B		
510	S2-SI-021-H-021	2C09	2BHA	B		
511	S2-SI-021-H-023	2C09	2BHA	B		
512	S2-SI-021-H-031	2C09	2BHA	B		
513	S2-SI-032-H-001	2C01	2BHA	B		
514	S2-SI-032-H-004	2C01	2BHA	B		
515	S2-SI-033-H-007	2C02	2BHA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

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SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
516	S2-SI-034-H-003	2C01	2BHA	B		
517	S2-SI-034-H-011	2C01	2BHA	B		
518	S2-SI-035-H-008	2C02	2BHA	B		
519	S2-SI-036-H-003	2C05	2BHA	B		
520	S2-SI-036-H-005	2C05	2BHA	B		
521	S2-SI-036-H-007	2C05	2BHA	B		
522	S2-SI-036-H-010	2C05	2BHA	B		
523	S2-SI-037-H-002	2C09	2BHA	B		
524	S2-SI-038-H-006	2C02	2BHA	B		
525	S2-SI-038-H-012	2C02	2BHA	B		
526	S2-SI-038-H-013	2C02	2BHA	B		
527	S2-SI-038-H-015	2C05	2BHA	B		
528	S2-SI-038-H-021	2C02	2BHA	B		
529	S2-SI-038-H-030	2C02	2BHA	B		
530	S2-SI-038-H-039	2C01	2BHA	B		
531	S2-SI-039-H-041	2C01	2BHA	B		
532	S2-SI-039-H-044	2C01	2BHA	B		
533	S2-SI-039-H-060	2C02	2BHA	B		
534	S2-SI-039-H-061	2C02	2BHA	B		
535	S2-SI-039-H-062	2C02	2BHA	B		
536	S2-SI-039-H-067	2C02	2BHA	B		
537	S2-SI-039-H-068	2C02	2BHA	B		
538	S2-SI-039-H-072	2C01	2BHA	B		
539	S2-RC-017-H-048	2C01	2BHA	B		
540	S2-SI-038-H-076	2C09	2BHA	B		
541	S2-SI-038-H-078	2C09	2BHA	B		
542	S2-SI-038-H-081	2C09	2BHA	B		
543	S2-SI-038-H-084	2C09	2BHA	B		
544	S2-SI-038-H-087	2C09	2BHA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
545	S2-SI-038-H-089	2C05	2BHA	B		
546	S2-SI-038-H-092	2C02	2BHA	B		
547	S2-SI-038-H-094	2C02	2BHA	B		
548	S2-SI-038-H-095	2C01	2BHA	B		
549	S2-SI-038-H-100	2C13	2BHA	B		
550	S2-SI-038-H-104	2C09	2BHA	B		
551	S2-SI-038-H-105	2C14	2BHA	B		
552	S2-SI-038-H-106	2C09	2BHA	B		
553	S2-SI-048-H-004	2C02	2BHA	B		
554	S2-SI-048-H-007	2C02	2BHA	B		
555	S2-SI-048-H-011	2C02	2BHA	B		
556	S2-SI-048-H-012	2C02	2BHA	B		
557	S2-SI-049-H-003	2C01	2BHA	B		
558	S2-SI-050-H-003	2C01	2BHA	B		
559	S2-SI-050-H-006	2C01	2BHA	B		
560	S2-SI-052-H-010	2C02	2BHA	B		
561	S2-SI-052-H-022	2C02	2BHA	B		
562	S2-SI-052-H-023	2C02	2BHA	B		
563	S2-SI-053-H-003	2C01	2BHA	B		
564	S2-SI-059-H-009	2C09	2BHA	C		
565	S2-SI-062-H-005	2C10	2BHA	C		
566	S2-SI-063-H-018	2C09	2BHA	C		
567	S2-SI-066-H-032	2C10	2BHA	C		
568	S2-SI-067-H-002	2C09	2BHA	C		
569	S2-SI-074-H-002	2C10	2BHA	C		
570	S2-SI-074-H-003	2C10	2BHA	C		
571	S2-SI-077-H-006	2C06	2BHA	C		
572	S2-SI-077-H-008	2C06	2BHA	C		
573	S2-SI-077-H-012	2C06	2BHA	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
574	S2-SI-108-H-003	2C02	2BHA	B		
575	S2-SI-109-H-002	2C02	2BHA	B		
576	S2-SI-109-H-005	2C02	2BHA	B		
577	S2-SI-132-H-002	2C02	2BHA	B		
578	S2-SI-139-H-005	2C05	2BHA	B		
579	S2-SI-139-H-012	2C05	2BHA	B		
580	S2-SI-139-H-015	2C01	2BHA	B		
581	S2-SI-139-H-031	2C01	2BHA	B		
582	S2-SI-145-H-003	2C05	2BHA	B		
583	S2-SI-145-H-005	2C05	2BHA	B		
584	S2-SI-146-H-004	2C09	2BHA	B		
585	S2-SI-149-H-010	2C09	2BHA	B		
586	S2-SI-149-H-011	2C09	2BHA	B		
587	S2-SI-149-H-013	2C09	2BHA	B		
588	S2-SI-149-H-015	2C09	2BHA	B		
589	S2-SI-149-H-018	2C09	2BHA	B		
590	S2-SI-164-H-005	2C09	2BHA	C		
591	S2-SI-065-H-028	2C06	2BHA	C		
592	S2-SI-077-H-019	2C06	2BHA	C		
593	S2-FS-107-H-002	2C11	2BHB	C		
594	S2-SI-107-H-006	2C11	2BHB	C		
595	S2-SI-107-H-007	2C07	2BHB	C		
596	S2-SI-080-H-011	2C03	2BHB	C		
597	S2-SI-080-H-013	2C02	2BHB	C		
598	S2-SI-080-H-028	2C02	2BHB	B		
599	S2-SI-080-H-030	2C02	2BHB	B		
600	S2-SI-080-H-033	2C02	2BHB	B		
601	S2-SI-080-H-047	2C07	2BHB	C		
602	S2-SI-080-H-050	2C07	2BHB	C		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
603	S2-SI-131-H-006	2C02	2BHB	B		
604	S2-SI-131-H-009	2C02	2BHB	B		
605	S2-SI-151-H-007	2C02	2BHB	B		
606	S2-SI-151-H-008	2C02	2BHB	B		
607	S2-SI-151-H-011	2C02	2BHB	B		
608	S2-SI-151-H-013	2C02	2BHB	B		
609	S2-CS-001-H-015	2C01	2BKA	B		
610	S2-CS-011-H-021	2C02	2BKA	B		
611	S2-CS-011-H-022	2C02	2BKA	B		
612	S2-CS-002-H-014	2C02	2BKA	B		
613	S2-CS-003-H-011	2C01	2BKA	B		
614	S2-CS-003-H-041	2C05	2BKA	B		
615	S2-CS-003-H-043	2C05	2BKA	B		
616	S2-CS-003-H-044	2C05	2BKA	B		
617	S2-CS-003-H-046	2C05	2BKA	B		
618	S2-CS-003-H-047	2C05	2BKA	B		
619	S2-CS-003-H-053	2C01	2BKA	B		
620	S2-CS-003-H-059	2C02	2BKA	B		
621	S2-CS-003-H-061	2C01	2BKA	B		
622	S2-CS-003-H-063	2C01	2BKA	B		
623	S2-CS-003-H-065	2C01	2BKA	B		
624	S2-CS-003-H-068	2C02	2BKA	B		
625	S2-CS-003-H-075	2C02	2BKA	B		
626	S2-CS-003-H-077	2C02	2BKA	B		
627	S2-CS-004-H-006	2C02	2BKA	B		
628	S2-CS-004-H-014	2C05	2BKA	B		
629	S2-CS-004-H-028	2C09	2BKA	B		
630	S2-CS-004-H-034	2C01	2BKA	B		
631	S2-CS-004-H-039	2C01	2BKA	B		

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TABLE 3.7-4b

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SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
632	S2-CS-004-H-043	2C01	2BKA	B		
633	S2-CS-004-H-048	2C09	2BKA	B		
634	S2-CS-004-H-053	2C15	2BKA	B		
635	S2-CS-004-H-054	2C13	2BKA	B		
636	S2-CS-004-H-056	2C01	2BKA	B		
637	S2-CS-004-H-058	2C02	2BKA	B		
638	S2-CS-004-H-062	2C02	2BKA	B		
639	S2-CS-004-H-074	2C02	2BKA	B		
640	S2-CS-004-H-079	2C02	2BKA	B		
641	S2-CS-004-H-080	2C01	2BKA	B		
642	S2-CS-018-H-002	2C02	2BKA	B		
643	S2-CS-027-H-014	2C02	2BKA	B		
644	S2-CS-027-H-015	2C02	2BKA	B		
645	S2-CS-027-H-020	2C02	2BKA	B		
646	S2-CS-027-H-022	2C02	2BKA	B		
647	S2-CS-028-H-007	2C02	2BKA	B		
648	S2-CS-032-H-005	2C02	2BKA	B		
649	S2-CS-032-H-006	2C02	2BKA	B		
650	S2-CS-032-H-008	2C02	2BKA	B		
651	S2-CS-033-H-004	2C02	2BKA	B		
652	S2-CS-041-H-023	2C09	2BKA	C		
653	S2-CS-047-H-017	2C05	2BKA	C		
654	S2-CS-047-H-022	2C05	2BKA	C		
655	S2-SI-004-H-032	2C02	2BKA	B		
656	S2-SI-004-H-034	2C02	2BKA	B		
657	S2-SI-114-H-001	2C02	2BKA	B		
658	S2-SI-115-H-117	2C02	2BKA	B		
659	S2-SI-115-H-020	2C02	2BKA	B		
660	S2-SI-117-H-001	2C02	2BKA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
661	S2-SI-117-H-002	2C02	2BKA	B		
662	S2-SI-119-H-002	2C02	2BKA	B		
663	S2-SI-119-H-003	2C02	2BKA	B		
664	S2-SI-119-H-004	2C02	2BKA	B		
665	S2-SI-119-H-007	2C02	2BKA	B		
666	S2-SI-119-H-008	2C92	2BKA	B		
667	S2-SI-114-H-008	2C02	2BKA	B		
668	S2-SI-115-H-025	2C02	2BKA	B		
669	S2-ST-015-H-004	2C05	2BMD	A		
670	S2-ST-015-H-005	2C06	2BMD	A		
671	S2-ST-015-H-012	2C06	2BMD	C		
672	S2-ST-015-H-013	2C06	2BMD	C		
673	S2-ST-015-H-014	2C06	2BMD	C		
674	S2-ST-106-H-009	2C09	2BMD	C		
675	S2-ST-106-H-010	2C05	2BMD	C		
676	S2-ST-106-H-011	2C05	2BMD	C		
677	S2-ST-106-H-012	2C05	2BMD	C		
678	S2-ST-106-H-013	2C05	2BMD	C		
679	S2-ST-106-H-014	2C05	2BMD	C		
680	S2-FS-007-H-003	2C09	2ECA	B		
681	S2-FS-007-H-005	2C09	2ECA	B		
682	S2-FS-007-H-007	2C09	2ECA	B		
683	S2-FS-007-H-017	2C05	2ECA	B		
684	S2-FS-007-H-018	2C05	2ECA	B		
685	S2-FS-007-H-022	2C09	2ECA	B		
686	S2-FS-025-H-005	2C13	2ECA	C		
687	S2-FS-030-H-002	2C13	2ECA	C		
688	S2-FS-139-H-001	2C09	2ECA	B		
689	S2-FS-139-H-002	2C09	2ECA	B		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

	SNUBBER NUMBER	AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
690	S2-FS-139-H-008	2C09	2ECA	B		
691	S2-FS-139-H-013	2C13	2ECA	B		
692	S2-SI-133-H-020	2C01	2ECA	B		
693	S2-SI-133-H-024	2C01	2ECA	B		
694	S2-SI-133-H-028	2C05	2ECA	B		
695	S2-SI-133-H-030	2C05	2ECA	B		
696	S2-SI-133-H-039	2C05	2ECA	B		
697	S2-SI-133-H-047	2C09	2ECA	B		
698	S2-SI-133-H-049	2C13	2ECA	B		
699	S2-FS-007-H-009	2C09	2ECA	B		
700	SA-CC-112-H-002	CA06	2EGA	C		
701	SA-CC-127-H-009	CA05	2EGA	C		
702	S2-CC-004-H-006	2C06	2EGA	A		
703	S2-CC-004-H-007	2C06	2EGA	A		
704	S2-CC-005-H-008	2C01	2EGA	C		
705	S2-CC-006-H-009	2C09	2EGA	A		
706	S2-CC-007-H-008	2C02	2EGA	B		
707	S2-CC-008-H-008	2C09	2EGA	A		
708	S2-CC-008-H-010	2C09	2EGA	A		
709	S2-CC-008-H-018	2C09	2EGA	A		
710	S2-CC-008-H-019	2C09	2EGA	A		
711	S2-CC-008-H-020	2C05	2EGA	A		
712	S2-CC-008-H-021	2C09	2EGA	A		
713	S2-CC-010-H-001	2C10	2EGA	A		
714	S2-CC-013-H-017	2C10	2EGA	A		
715	S2-CC-014-H-002	2C05	2EGA	A		
716	S2-CC-014-H-003	2C05	2EGA	A		
717	S2-CC-014-H-005	2C05	2EGA	A		
718	S2-CC-014-H-006	2C05	2EGA	A		

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TABLE 3.7-4b

SAFETY RELATED MECHANICAL SNUBBERS*

SNUBBER NUMBER		AREA	SUS	S/D RAD**	ACCESS OR INACCESS	DIFF. TO REMOVE
719	S2-CC-146-H-005	2C01	2EGA	C		
720	S2-CC-149-H-002	2C01	2EGA	C		
721	S2-CC-266-H-025	2C13	2EGA	C		
722	S2-SC-001-H-001	2C01	2EPA	C		
723	S2-SC-001-H-016	2S01	2EPA	C		
724	S2-SC-002-H-005	2S01	2EPA	C		
725	S2-SC-002-H-008	2S02	2EPA	C		
726	S2-SC-003-H-011	2S01	2EPA	C		
727	S2-ST-016-H-015	2C05	2BMD	C		
728	S2-ST-016-H-016	2C05	2BMD	C		
729	S2-CC-292-H-020	2C13	2EGA	C		
730	SA-CH-390-H-005	CA10	2GKA	C		
731	S2-CC-275-H-008	2C10	2GNF	C		
732	S2-CC-275-H-009	2C10	2GNF	C		
733	S2-CC-287-H-009	2C09	2GNF	C		
734	S2-CC-287-H-011	2C09	2GNF	C		
735	S2-CB-005-H-001	2C09	2GNF	C		
736	S2-CB-006-H-001	2C05	2GNF	C		
737	S2-GR-052-H-004	2C09	2HAA	C		
738	S2-GR-065-H-010	2C09	2HAA	C		
739	S2-GR-066-H-022	CA05	2HAA	A		
740	S3-GR-066-H-012	CA06	2HAA	A		

— There are 216 CEA snubbers To be added To this list —

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3/4.7.~~8~~ 7 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.~~8~~ 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 either:

   1. Decontaminate and repair the sealed source, or

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

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

SURVEILLANCE REQUIREMENTS

4.7.~~8~~ 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.~~8~~ 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:

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

   2. In any form other than gas.

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SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

c. Startup sources and fission detectors - Each sealed startup source and fission detector shall be tested within 31 days prior to being subjected to core flux or installed in the core and following repair or maintenance to the source or detector.

4.7.~~8~~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.

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3/4.7.~~9~~8 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.~~9~~8.1 The fire suppression water system shall be OPERABLE with:

a. Two electric motor-driven fire pumps, each with a capacity of 1500 gpm and~~/or~~ one diesel-driven fire pump ~~on each~~ with a capacity of 2500 gpm, 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 each water supply and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrant curb valves, the first valve upstream of the water flow alarm device on each sprinkler, hose standpipe, or spray system riser required to be OPERABLE per Specifications 3.7.~~11~~8.2, 3.7.~~11~~8.5 and 3.7.~~11~~8.6.

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

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THIS [illegible] RECEIPT OF
INFO[illegible] APPLICANT

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.~~9~~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 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 performance of the fire pumps as follows: ~~that each pump develops at least (2500) gpm at a system head of (250) feet;~~

   a. Diesel engine driven pump develops at least 2500 gpm at a system head of 283 feet.

   b. Electric motor driven pumps each develop at least 1500 gpm at a system head of 289 feet.

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

3. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 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.~~8~~8.1.2 The fire pump diesel engine shall be demonstrated OPERABLE:

a. At least once per 31 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.~~8~~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.

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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 deterioration, and

   2. The battery-to-battery and terminal connections are clean, tight, free of corrosion, and coated with anti-corrosion material.

EX

PLANT SYSTEMS

SPRAY AND/OR SPRINKLER SYSTEMS

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LIMITING CONDITION FOR OPERATION

3.7.~~9~~8.2 The spray and/or sprinkler systems listed in Table 3.7-5 shall be OPERABLE.

APPLICABILITY: Whenever equipment protected by the spray/sprinkler system is required to be OPERABLE.

ACTION:

a. With one or more of the above required spray and/or sprinkler systems inoperable, within one hour establish a continuous fire watch with backup fire suppression equipment for those areas ~~in which~~ outside containment redundant systems or components could be damaged; for other areas outside containment, 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.~~9~~8.2 Each of the above required spray and/or sprinkler systems shall be demonstrated OPERABLE:

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

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

Note: RCP oil collection system has eliminated need for in-containment sprinklers to mitigate fires resulting from RCP oil leaks.

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TABLE 3.7-5

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
Charcoal Filter A-358	Containment	1	Deluge-Water Spray
Emergency A.C. Unit-Train A	Fuel Handling Bldg	1	Deluge-Water Spray
Emergency A.C. Unit-Train B	Fuel Handling Bldg	1	Deluge-Water Spray
Charcoal Filter E-419	Control Bldg	1	Deluge-Water Spray
Charcoal Filter A-206	Control Bldg	1	Deluge-Water Spray
Diesel Generator	DG Bldg	2	Pre-Action Sprinkler

update to reflect FHA Q015.60

* one half of these Systems are designated Unit 3 But are required to be operable for Unit 2 operation

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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~~ wet pipe spray and sprinkler headers to verify their integrity, and

3. By a visual inspection of each ~~nozzle's spray area~~ spray/sprinkler head to verify ~~the~~ its 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.

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PLANT SYSTEMS

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.~~9~~8.3 The fire hose stations shown in Table 3.7-6 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the fire hose stations is required to be OPERABLE.

ALARA

ACTION:

a. With one or more of the fire hose stations outside containment shown in Table 3.7-6 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 suppresion; 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

ALARA

4.7.~~9~~8.3 Each of the fire hose stations shown in Table 3.7-6 shall be demonstrated OPERABLE:

Note: containment is a controlled access area

a. At least once per 31 days by visual inspection of the stations outside containment and 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 inside containment or otherwise 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 3.7-8

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

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No hydrants are used for either primary or backup protection for safety related equipment!!

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.9.4 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-7 shall be OPERABLE.

APPLICABILITY: Whenever equipment in the areas protected by the yard fire hydrants is required to be OPERABLE.

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-7 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

4.7.9.4 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-7 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.

TABLE 3.7-7

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION*	HYDRANT NUMBER

No hydrants used for protection of safety-related equipment

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THIS ... RECEIPT OF ... UNIT

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

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PLANT SYSTEMS

3/4.7.~~10~~ 9 FIRE BARRIER PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.7.~~10~~ 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.~~10~~ 9 Each of the above required fire barrier penetrations shall be verified to be functional:

a. For fire doors and fire dampers, at ~~At~~ least once per 18 months by a visual inspection.

c. ~~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).

b. For fire barrier penetrations other than fire doors and fire dampers, at least once per 18 months by a visual inspection of a representative sample of at least 10% of such fire barrier penetrations, selected on a rotating basis.

Req'd due to large number (>10,000) of such penetration barriers

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3/4.7.11 AREA TEMPERATURE MONITORING

LIMITING CONDITION FOR OPERATION

3.7.11 The temperature of each area shown in Table 3.7-8 shall be maintained within the limits indicated in Table 3.7-8.

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 3.7-8 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

4.7.11 The temperature in each of the areas of Specification 3.7.11 shall be determined to be within its limit at least once per 24 hours.

Not required for SO 2/3 due to coastal location, regional climate and qualification of equipment (see FSAR sections 2.3 and 3.11)

NOTE!

TABLE 3.7-8

AREA TEMPERATURE MONITORING

AREA	TEMPERATURE LIMIT (°F)
1.	
2.	
3.	
4.	
5.	

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not req'd for SO 2/3

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3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

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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 day ~~and~~ fuel tank 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.

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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 fuel tank,

2. Verifying the fuel level in the fuel storage tank,

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,

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Notes:
1. LOV and SIAS start contacts in parallel so nothing gained by testing coincident signals (simulated SIAS or LOV)
2. Cannot simulate starts without
a. dropping the bus
b. using jumpers
c. challenging offsite power during plant operation.

4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in less than or equal to 10 seconds. The generator voltage and frequency shall be 4360 ± 436 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~~ manual start signal~~s with startup on each signal verified at least once per 184 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 kw in less than or equal to 77 seconds, and operates at full load 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 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day tank.

c. At least once per 92 days and from new fuel oil prior to addition to the storage tanks by verifying that a sample obtained in accordance with ASTM-D270-1975 has a water and sediment content of less than or equal to .05 volume percent and a kinematic viscosity @40°C of greater than or equal to 1.3 but less than or equal to 2.4 when tested in accordance with ASTM-D975-77, and an impurity level of less than 2 mg of insolubles per 100 ml. when tested in accordance with ASTM-D2274-70.

d. At least once per 18 months during shutdown by:

1. Subjecting the diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service.

2. Verifying the generator capability to reject a load of greater than or equal to 655.7 kw for Generator G002 and [redacted] 655.7 kw for Generator G003 while maintaining voltage at 4360 ± 440 volts and frequency at 60 ± 5.25 Hz.

due to add^n of 3^rd AFW pump

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SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the generator capability to reject a load of 4700 kw without tripping. The generator voltage shall not exceed (4784) volts during and following the load rejection.

BPC verify

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~~ permanently connected loads. After energization, the steady state voltage and frequency of the emergency busses shall be maintained at 4360 ± 1090 volts and 60 ~~± 3~~ +3 -1.2 Hz during this test.

no monolithic sequencer; no auto-connected S/D loads; 480V loadcenters only auto-energized and most of these loads are not continuous

5. Verifying that on an SIAS 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 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~~ SIAS 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,

Note: no monolithic sequencer; so load sequencing independent of bus power source

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no monolithic sequencer

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 4360 ± 1090 volts and 60 [redacted] Hz during this test. +3 -1.2

necessary for adequate HPSI sim. BE LOCA performance

c) Verifying that all automatic diesel generator trips, except engine overspeed [redacted] generator differential and low lube oil, are automatically bypassed upon ~~loss of voltage on the emergency bus concurrent with~~ a safety injection actuation test signal.

actual logic

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 5170 kw and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal to 4700 kw. Within 5 minutes after completing this 24 hour test, perform Specification 4.8.1.1.2.d.4. The generator voltage and freqency shall be 4360 ± 1090 volts and 60 ± 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.

no 2000 hr rating

~~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 tank of each diesel via the installed cross connection lines.

[redacted]

SURVEILLANCE REQUIREMENTS (Continued)

~~13.~~ 12. Verifying that the automatic load sequence ~~timer is~~ timers are OPERABLE with the interval between each load block within ± 10% of its design interval.

~~14.~~ 13. Verifying that ~~the following diesel non-critical generator lockout features~~ lockout relay (K23) prevents diesel generator starting when ~~the diesel generator is~~ actuated.

~~a. Jacket coolant high temperature~~

~~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~~

e. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting the diesel generators simultaneously, during shutdown, and verifying that the diesel generators accelerate to at least 900 rpm in less than or equal to 10 seconds.

f. At least once per 10 years by:

1. Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank using a sodium hypoclorite solution, and

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code at a test pressure equal to 110 percent of the system design pressure.

4.8.1.1.3 Reports - All diesel generator failures, valid or non-valid, shall be reported to the Commission pursuant to Specification 6.9.1. Reports of diesel generator failures shall include the information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977. If the number of failures in the last 100 valid tests (on a per nuclear unit basis) is greater than or equal to 7, the report shall be supplemented to include the additional information recommended in Regulatory Position C.3.b of Regulatory Guide 1.108, Revision 1, August 1977.

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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.

[redacted]

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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 fuel tanks containing a minimum volume of 325 gallons of fuel,

   2. A fuel storage system containing a minimum volume of 48,736 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.

8

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 and inverters shall be OPERABLE and energized.

later

BPC to provide

later

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 energized from Inverter # ____ connected to D.C. Bus Train ____.*
120	volt A.C. Vital Bus # 2YV2 energized from Inverter # ____ connected to D.C. Bus Train ____.*
120	volt A.C. Vital Bus # 2YV3 energized from Inverter # ____ connected to D.C. Bus Train ____.*
120	volt A.C. Vital Bus # 2YV4 energized from Inverter # ____ connected to D.C. Bus Train ____.*

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses and inverters OPERABLE or energized, restore the inoperable busses and inverters to OPERABLE and energized 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 and inverters shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated voltage on the bus.

*An inverter may be disconnected from its D.C. source for up to 24 hours for the purpose of performing an equalizing charge on its associated battery bank provided (1) its vital bus is OPERABLE and energized, and (2) the vital busses associated with the other battery banks are OPERABLE and energized.

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses and inverters shall be OPERABLE and energized:

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Bus

2 - 120 volt A.C. Vital Busses energized from their respective inverters connected to their respective bus train.

APPLICABILITY: MODES 5 and 6

ACTION:

With less than the above complement of A.C. busses and inverters OPERABLE and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses and inverters shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

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D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.3 The following D.C. bus trains shall be OPERABLE and energized:

Channel "A" And ~~TRAIN "A" consisting of (250/125)-volt D.C. bus No. 1, (250/125)-volt lead battery bank No. 1, and a full capacity charger.~~

Channel "B" And

Channel "C" And

Channel "D" ~~TRAIN "B" consisting of (250/125)-volt D.C. bus No. 2, (250/125)-volt lead 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 or not energized, restore the inoperable bus train to OPERABLE and energized 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.

~~This specification is intended for use on plants with two divisions of D.C. power only. Modifications may be necessary, on a plant-unique basis, to accommodate different designs.~~

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized with tie breakers open between redundant busses at least once per 7 days by verifying correct breaker alignment, indicated power availability from the charger and battery, and voltage on the bus 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:

1. Verifying that the parameters in Table 4.8-2 meet the Category A limits, and

2. Verifying total battery terminal voltage is greater than or equal to ~~(258/~~129~~)~~-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge (battery terminal voltage below ~~(220/~~110) volts), or battery overcharge (battery terminal voltage above ~~(300/~~150) volts), by:

1. Verifying that the parameters in Table 4.8-2 meet the Category B limits,

2. Verifying there is no visible corrosion at either terminals or connectors, if corrosion is visible, verify that ~~or~~ the connection resistance of these items is less than (150 x 10-6) ohms, and

3. Verifying that the average electrolyte temperature of (a representative number) of connected cells is above ~~(60 F)~~ ~~70°F~~ 60°F

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

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to (150 x 10-6) ohms, and

4. The battery charger will supply at least ~~(400)~~ 300 amperes at (125~~/250)~~ volts for at least ~~(8)~~ 12 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	≥ (1.200)(b)	≥ (1.195)	Not more than .020 below the average of all connected cells
		Average of all connected cells > ~~(1.205)~~ 1.200	Average of all connected cells ≥ (1.195)(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than (2) amps.
(c) Corrected for average electrolyte temperature.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all parameter(s) are restored to within limits withint the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that they are within their allowable values and provided the parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

~~Numbers in paretheses assume a manufacturer's recommended full charge specific gravity of 1.215.~~

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ELECTRICAL POWER SYSTEMS

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 OPERABLE and energized:

1 - 125-volt D.C. bus, and

1 - 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 and energized, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and voltage on the bus with an overall voltage of greater than or equal to 125 volts.

4.8.2.4.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

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3/4.8.3 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

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.

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SURVEILLANCE REQUIREMENTS (Continued)

(c) For each circuit breaker found inoperable during these functional tests, an additional representative sample of at least 10% of all the circuit breakers of the inoperable type shall also be functionally tested until no more failures are found or all circuit breakers of that type have been functionally tested.

2. By selecting and functionally testing a representative sample of at least 10% of each type of lower voltage circuit breakers. Circuit breakers selected for functional testing shall be selected on a rotating basis. 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

CONTAINMENT PENETRATION CONDUCTOR

OVERCURRENT PROTECTIVE DEVICES

Device Number and Location	Trip Setpoint (Amperes)	Response Time (sec/cycles)	System Powered
1. 6900 VAC			Reactor Coolant pump
(Primary breaker)			1
(Back-up breaker)			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			
" "			

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ELECTRICAL POWER SYSTEMS

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION ~~AND/OR~~ BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.3.2 The thermal overload protection relay contact, integral with the motor starter, of each valve listed in Table 3.8.2 is permanently bypassed.

APPLICABILITY: Whenever the motor operated valve is required to be OPERABLE.

ACTION:

With one or more of the thermal overload protection 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 relay CONTACT bypass devices shall be visually inspected to determine that the protective devices are installed.

1. After maintenance on ~~this~~ the motor starter.

2. At least once per 18 months.

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TABLE 3.8-2

MOTOR OPERATED VALVES THERMAL OVERLOAD PROTECTION AND/OR BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
HV-9339	Shutdown cooling flow ~~then~~ from reactor coolant loop 2	Permanently Bypassed
HV-9340	SI tank T008 to reactor coolant loop 1A	Permanently Bypassed
HV-9370	SI tank T010 to reactor coolant loop 2B	Permanently Bypassed
HV-9347	SI pump minimum recirculation	Permanently Bypassed
HV-9322	LPSI to reactor coolant loop 1A	Permanently Bypassed
HV-9331	LPSI to reactor coolant loop 2B	Permanently Bypassed
HV-9348	SI pump minimum recirculation	Permanently Bypassed
HV-9323	HPSI to reactor coolant loop 1A	Permanently Bypassed
HV-9332	HPSI to reactor coolant loop 2B	Permanently Bypassed
HV-9217	RCP bleed off to volume control tank	Permanently Bypassed
HV-9326	HPSI to reactor coolant loop 1B	Permanently Bypassed
HV-9329	HPSI to reactor coolant loop 2A	Permanently Bypassed
HV-7258	Containment ~~150~~ iso SI tank ~~vent~~	Permanently Bypassed
HV-0508	Reactor coolant hot leg sample containment ~~150~~ iso	Permanently Bypassed
HV-0517	Reactor coolant hot leg sample containment ~~150~~ iso	Permanently Bypassed
HV-9368	Shutdown HX to containment spray	Permanently Bypassed
HV-0510	Pressurizer vapor sample containment ~~150~~ iso	Permanently Bypassed
HV-0512	Pressurizer surge line liquid sample containment ~~150~~ iso	Permanently Bypassed
HV-9950	Containment purge inlet	Permanently Bypassed
HV-9417	Hydrogen purge exhaust inlet	Permanently Bypassed
HV-9946	Hydrogen purge supply discharge	Permanently Bypassed
HV-9302	Containment emergency sump outlet	Permanently Bypassed
HV-9304	Containment emergency sump outlet	Permanently Bypassed

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TABLE 3.8-1 (CONT'D)

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
HV-6211	CCW to containment	Permanently Bypassed
HV-6368	CCW to emergency cooling unit	Permanently Bypassed
HV-6369	CCW to emergency cooling unit	Permanently Bypassed
HV-6216	CCW from containment	Permanently Bypassed
HV-6372	CCW to emergency cooling unit	Permanently Bypassed
HV-6373	CCW to emergency cooling unit	Permanently Bypassed
HV-9900	Containment normal cooling supply ~~150~~ iso	Permanently Bypassed
HV-9971	Containment normal cooling return ~~150~~ iso	Permanently Bypassed
LV-~~0237~~ 0227C	Boric Acid makeup control	Permanently Bypassed
HV-4713	Aux. F.W. to steam generator control	Permanently Bypassed
HV-9334	SI tank drain to refueling water tank	Permanently Bypassed
HV-9350	SI tank T007 to reactor coolant loop 1B	Permanently Bypassed
HV-9360	SI tank T009 to reactor coolant loop 2A	Permanently Bypassed
HV-9325	LPSI to reactor coolant loop 1B	Permanently Bypassed
HV-9328	LPSI to reactor coolant loop 2B	Permanently Bypassed
HV-9201	Aux. spray to pressurize	Permanently Bypassed
HV-9327	HPSI to reactor coolant loop 1B	Permanently Bypassed
HV-9330	HPSI to reactor coolant loop 2A	Permanently Bypassed
HV-6223	CCW Non-Crit Containment inlet 150	Permanently Bypassed
HV-9324	HPSI to reactor coolant loop 1A	Permanently Bypassed
HV-9333	HPSI to reactor coolant loop 2B	Permanently Bypassed
HV-9337	Shutdown coolant flow from reactor coolant loop 2	Permanently Bypassed
HV-0516	Reactor coolant drain tank sample containment 150	Permanently Bypassed
V-7512	Containment ~~150~~ iso reactor coolant drain to R.W. system	Permanently Bypassed

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TABLE 3.8-1 (CONT'D)

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
HV-9367	Shutdown HX to containment spray header	Permanently Bypassed
HV-0514	Quench tank vapor sample containment 150	Permanently Bypassed
HV-5803	Containment sump to R.W. sump	Permanently Bypassed
HV-9949	Containment purge outlet	Permanently Bypassed
HV-9303	Containment emergency sump outlet	Permanently Bypassed
HV-9305	Containment emergency sump outlet	Permanently Bypassed
HV-6366	CCW to emergency cooling unit	Permanently Bypassed
HV-6367	CCW to emergency cooling unit	Permanently Bypassed
HV-6236	CCW Non-crit. containment ~~150~~ iso valve	Permanently Bypassed
HV-6370	CCW to emergency cooling unit	Permanently Bypassed
HV-6371	CCW to emergency cooling unit	Permanently Bypassed
HV-8150	Reactor aux. shutdown cooling HX outlet	Permanently Bypassed
HV-8152	Reactor aux shutdown cooling HX inlet	Permanently Bypassed
HV-9306	SI pump mini-flow	Permanently Bypassed
HV-9307	SI pump mini-flow	Permanently Bypassed
HV-9247	Boric acid pumps to charging pump suction	Permanently Bypassed
HV-9379	Shutdown cooling flow to LPSI	Permanently Bypassed
HV-9353	Shutdown cooling warm up valve	Permanently Bypassed
HV-9420	HPSI to reactor coolant loop 2	Permanently Bypassed
HV-6497	Saltwater from CCW HX	Permanently Bypassed
HV-9300	Refueling water tank east outlet	Permanently Bypassed
HV-5686	Firewater to containment ~~150~~ iso	Permanently Bypassed
HV-0227B	Valume control tank drain return	Permanently Bypassed
HV-9240	Boric acid make uptank T072 to charging pump suction	Permanently Bypassed

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TABLE 3.8-1 (CONT'D)

VALVE NUMBER	FUNCTION	BYPASS DEVICE (YES/NO)
HV-9336	Shutdown cooling flow to LPSI pump suction	Permanently Bypassed
HV-9359	Shutdown cooling warm up valve	Permanently Bypassed
HV-9301	Refueling water tank west outlet	Permanently Bypassed
HV-6495	Saltwater from CCW HX	Permanently Bypassed
TV-9267	Reactor coolant regenerative HX ~~150~~ iso	Permanently Bypassed
HV-9434	HPSI to reactor coolant loop 1 hot leg	Permanently Bypassed
HV-8151	Reactor aux. shutdown cooling HX outlet	Permanently Bypassed
HV-8153	Reactor aux shutdown cooling HX inlet	Permanently Bypassed
HV-4712	Aux F.W. Steam gen. control	Permanently Bypassed

Note: need special excepts for initial (dry) fueling

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of 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 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 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 1720 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 fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

Note: need special exceptns for initial (dry) fueling

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3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head closure bolts less than fully tensioned or with the head removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, ~~which includes~~ including a 1% delta k/k conservative allowance for uncertainties, or

b. A boron concentration of greater than or equal to 1720 ppm, ~~which includes~~ including 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 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 1720 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 fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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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.

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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.

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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.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

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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 4.6.3.2.

3/4.9.5 COMMUNICATIONS

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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.

3/4.9.6 REFUELING MACHINE

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LIMITING CONDITION FOR OPERATION

3.9.6 The refueling machine shall be used for movement of CEAs or fuel assemblies and shall be OPERABLE with:

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a. A minimum capacity of (2750) pounds, and

b. An 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 the refueling machine OPERABILITY not satisfied, suspend 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 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 test of at least (2750) pounds and demonstrating an automatic load cut off when the refueling machine load exceeds (2700) pounds.

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REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of ____ pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of ____ pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

There is no crane over the spent fuel racks, and due to building configurations cask handling crane cannot either.

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3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of ____ pounds shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of ____ pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane use and at least once per 7 days thereafter during crane operation.

There is no crane over the fuel racks, [illegible] building [illegible] handling cask [illegible] crane cannot [illegible] either.

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3/4.9.~~8~~ 7 SHUTDOWN COOLING AND COOLANT CIRCULATION

ALL WATER LEVELS

LIMITING CONDITION FOR OPERATION

3.9.~~8~~ 7.1 At least one shutdown cooling ~~loop~~ train shall be in operation.

correct SO 2/3 terminology

APPLICABILITY: MODE 6

ACTION:

a. With less than one shutdown cooling ~~loop~~ train 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.

N/A to Mode 6

b. The shutdown cooling ~~loop~~ train may be removed from operation for up to 1 hour per ~~8~~ 4 hour period ~~during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.~~ provided (1) no operations are permitted that would cause dilution of the RCS boron concentration and (2) core outlet temperature is maintained 10°F below saturation temperature.

arbitrary; some limit required to preclude development of pool/vessel stratification

see spec 3/4.1.3

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) gpm at least once per 12 hours.

C-E to verify

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3/4.9.~~8~~7 SHUTDOWN COOLING 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) gpm at least once per 12 hours.

C-E to verify

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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 shutdown cooling loops ~~sall~~ shall be determined OPERABLE per Specification 4.0.5.

*The normal or emergency power source may be inoperable for each shutdown cooling loop.

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REFUELING OPERATIONS

3/4.9.~~11~~ 10 WATER LEVEL-STORAGE POOL

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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.

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3/4.9.~~12~~ 11 FUEL HANDLING BUILDING EMERGENCY ~~VENTILATION~~ AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.~~12~~ 11 Two independent fuel handling building emergency ~~ventilation~~ air cleanup systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building emergency ~~ventilation~~ air cleanup system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel handling building emergency ~~ventilation~~ air cleanup system is in operation and ~~discharging~~ recirculating through at least one train of HEPA filters and charcoal adsorbers.

recirculation cleanup, not once-through

b. With no fuel handling building emergency ~~ventilation~~ air cleanup system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel handling building emergency ~~ventilation~~ 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 fuel handling building emergency ~~ventilation~~ air cleanup systems shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters on 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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3/4.9.~~12~~ 11 FUEL HANDLING BUILDING EMERGENCY ~~VENTILATION~~ AIR CLEANUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.~~12~~ 11 Two independent fuel handling building emergency ~~ventilation~~ air cleanup systems shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one fuel handling building emergency ~~ventilation~~ air cleanup system inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE fuel handling building emergency ~~ventilation~~ air cleanup system is in operation at least 10 minutes during each 24 hours and ~~discharging~~ recirculating through at least one train of HEPA filters and charcoal adsorbers.

recirculation cleanup, not once-through

b. With no fuel handling building emergency ~~ventilation~~ air cleanup system OPERABLE, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one fuel handling building emergency ~~ventilation~~ 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 fuel handling building emergency ~~ventilation~~ air cleanup systems shall be demonstrated OPERABLE:

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

Note: operation of normal system is required to maintain FHB habitability (humidity, temperature, etc); operation of cleanup system isolates this.

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SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system recirculating ~~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. to the facility vent

SO2/3 have recirculation cleanup, not filtered exhaust. ∴ only path to vent is exhaust damper leakage

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%.

2 ~~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.

3 ~~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 7.3 ~~(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~~ isolates normal ventilation and starts recirculation through the HEPA filters and charcoal adsorber banks.

recirc system, not once-through

not normally operating ∴ not "already operating"

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SURVEILLANCE REQUIREMENTS (Continued)

SO2/3 have recirculation cleanup, not filtered exhaust. only path to vent is exhaust damper leakage

1. Verifying that with the system recirculating 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. to the facility vent

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%.

2 ~~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.

3 ~~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 7.3 ~~(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. isolates normal ventilation and starts recirculation

recirc system, not once-through

not now

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RECEIPT OF APPLICANT

SURVEILLANCE REQUIREMENTS (Continued)

N/A (recirc sys.)

~~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~~ 3. Verifying that the heaters dissipate ~~33 ± 1.7 kw~~ 28.4 ±1.5 kw for E464, 32.3 ±1.7 kw for E465, and 3.8 ±0.2 kw for E652 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 tbe system at a flow rate of 12925 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%.

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3/4.10 SPECIAL TEST EXCEPTIONS

3/4.10.1 SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of 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:

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 greater than or equal to 1720 ppm boron or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all full length CEAs fully 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 greater than or equal to 1720 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 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 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:

3.1.3.2

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 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 the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 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.

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 requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

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3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

Moderator temperature Coefficient,

To spell out item 3.1.1.3

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, 3.2.2, 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:

3.1.3.2

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.5, 3.1.3.6, 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 the requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 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.

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 requirements of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 or the Minimum Channels OPERABLE requirement of Functional Unit 15 of Table 3.3-1 are suspended.

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3/4.10.3 REACTOR COOLANT LOOPS

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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 logarithmic and linear power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

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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 logarithmic and linear power level neutron flux monitoring channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup and PHYSICS TESTS.

3/4.10.4 CENTER CEA MISALIGNMENT

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LIMITING CONDITION FOR OPERATION

3.10.4 The requirements of Specifications 3.1.3.1 and 3.1.3.6 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 3.1.3.6 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 3.1.3.6 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 3.1.3.6 are suspended.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

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LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-4) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1. The results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance with Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

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TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[d]	P Each Batch	P Each Batch	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma emitters)	1×10^{-5}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
			P-32	1×10^{-6}
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[e]	Grab Sample ~~Continuous~~[c]	W Composite[c]	Principal Gamma Emitters[f]	5×10^{-7}
			I-131	1×10^{-6}
to reflect 502/3 system	M Grab Sample	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Grab Sample ~~Continuous~~[c]	M Composite[c]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
			P-32	1×10^{-6}
	Grab Sample ~~Continuous~~[c]	Q Composite[c]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}

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TABLE 4.11-1 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and Δt shall be used in the calculation.

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TABLE 4.11-1 (Continued)

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TABLE NOTATION

b. A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be throughly mixed in order for the composite sample to be representative of the effluent release.

d. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed, by a method described in the ODCM, to assure representative sampling.

e. A continuous release is the discharge of liquid wastes of a nondiscrete volume; e.g., from a volume of system that has an input flow during the continuous release.

f. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

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RADIOACTIVE EFFLUENTS

DOSE

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LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to an individual from radioactive materials in liquid effluents released, from each reactor unit, from the site (see Figure 5.1-4) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrem to the total body and to less than or equal to 5 mrem to any organ, and

b. During any calendar year to less than or equal to 3 mrem to the total body and to less than or equal to 10 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose or dose commitment to an individual from these releases is within 3 mrem to the total body and 10 mrem to any organ. ~~(This Special Report shall also include (1) the results of radiological analyses of the drinking water source and (2) the radiological impact on finished drinking water supplies with regard to the requirements of 40 CFR 141, Safe Drinking Water Act.*)~~

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents shall be determined in accordance with the ODCM at least once per 31 days.

~~*Applicable only if drinking water supply is taken from the receiving water body.~~

no d.w. from Pacific Ocean!

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site (see Figure 5.1-4) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 15 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.

~~RADIOACTIVE EFFLUENTS~~

~~LIQUID HOLDUP TANKS*~~

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N/A

E

LIMITING CONDITION FOR OPERATION Delete

3.11.1.4 The quantity of radioactive material contained in each of the following tanks shall be limited to less than or equal to ____ curies excluding tritium and dissolved or entrained noble gases.

a. ________
b. ________
c. ________
d. Outside temporary tank

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any of the above listed tanks exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.1.4 The quantity of radioactive material contained in each of the above listed tanks shall be determined to be within the above limit by analyzing a representative sample of the tank's contents at least once per 7 days when radioactive materials are being added to the tank.

*Tanks included in this Specification are those outdoor tanks that are not surrounded by liners, dikes, or walls capable of holding the tank contents and that do not have tank overflows and surrounding area drains connected to the liquid radwaste treatment system.

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3/4.11.2 GASEOUS EFFLUENTS

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-3) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrem/yr to the total body and less than or equal to 3000 mrem/yr to the skin, and

b. For all radioiodines and for all radioactive materials in particulate form and radionuclides (other than noble gases) with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM.

4.11.2.1.2 The dose rate due to radioactive materials, other than noble gases, in gaseous effluents shall be determined to be within the above limits in accordance with the methods and procedures of the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[g]	1×10^{-4}
B. Containment Purge	P Each Purge[b] Grab Sample	P Each Purge[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
C. ~~(List other release points where gaseous effluents are discharged from the facility)~~ Condenser Air Ejector Plant Vent Stack	M[b,c,e] Grab Sample	M[b]	Principal Gamma Emitters[g]	1×10^{-4}
			H-3	1×10^{-6}
D. ~~All~~ Release Types as listed in ~~A,~~ B And C above.	Continuous[f]	W[d] Charcoal Sample	I-131	1×10^{-12}
			I-133	1×10^{-10}
	Continuous[f]	W[d] Particulate Sample	Principal Gamma Emitters[g] (I-131, Others)	1×10^{-11}
	Continuous[f]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[f]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[f]	Noble Gas Monitor	Noble Gases Gross Beta & Gamma	1×10^{-6}

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TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$\text{LLD} = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above (as microcurie per unit mass or volume),

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

E is the counting efficiency (as counts per transformation),

V is the sample size (in units of mass or volume),

2.22×10^6 is the number of transformations per minute per microcurie,

Y is the fractional radiochemical yield (when applicable),

λ is the radioactive decay constant for the particular radionuclide, and

Δt is the elapsed time between midpoint of sample collection and time of counting (for plant effluents, not environmental samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. Typical values of E, V, Y, and Δt shall be used in the calculation.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Analyses shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.

c. Tritium grab samples shall be taken at least once per 24 hours when the refueling canal is flooded.

d. Samples shall be changed at least once per 7 days and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15 percent of RATED THERMAL POWER in one hour and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10.

e. Tritium grab samples shall be taken at least once per 7 days from the ventilation exhaust from the spent fuel pool area, whenever spent fuel is in the spent fuel pool.

f. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

g. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

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LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

~~(The dose design objectives shall be reduced based on predicted noble gas releases from the turbine building if effluent sampling is not provided. The dose design objectives shall also be reduced based on expected public occupancy of areas, e.g., beaches and visitor centers within the site boundary.)~~

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases of radioactive noble gases in gaseous effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose is within (10) mrad for gamma radiation and (20) mrad for beta radiation.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

DOSE - RADIOIODINES, RADIOACTIVE MATER~~IALS IN~~ PARTICULATE FORM, AND RADIONUCLIDES OTHER THAN NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to an individual from radioiodines and radioactive materials in particulate form, and radionuclides (other than noble gases) with half-lives greater than 8 days in gaseous effluents released, from each reactor unit, from the site (see Figure 5.1-3) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ and,

b. During any calendar year: Less than or equal to 15 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioiodines, radioactive materials in particulate form, or radionuclides (other than noble gases) with half lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit and defines the corrective actions to be taken to reduce the releases of radioiodines and radioactive materials in particulate form, and radionuclides (other than nobles gases) with half-lives greater than 8 days in gaseous effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose or dose commitment to an individual from these releases is within (15) mrem to any organ.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Dose Calculations Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the ODCM at least once per 31 days.

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THIS ... RECEIPT OF ... APPLICANT

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site (see Figure 5.1-3), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-3) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site shall be projected at least once per 31 days, in accordance with the ODCM.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 15 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

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N/A

E

~~RADIOACTIVE EFFLUENTS~~

Delete

~~EXPLOSIVE GAS MIXTURE (Systems designed to withstand a hydrogen explosion)~~

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be limited to less than or equal to 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen or oxygen in the waste gas holdup system exceeding the limit, restore the concentration to within the limit within 48 hours.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentration of hydrogen or oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

[redacted]

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EXPLOSIVE GAS MIXTURE ~~(Systems not designed to withstand a hydrogen explosion)~~

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of hydrogen and/or oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of hydrogen and/or oxygen in the waste gas holdup system greater than 2% by volume but less than or equal to 4% by volume, restore the concentration of hydrogen and/or oxygen to within the limit within 48 hours.

b. With the concentration of hydrogen and/or oxygen in the waste gas holdup system greater than 4% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of hydrogen and/or oxygen to less than or equal to 2% within one hour.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5 The concentrations of hydrogen and/or oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and/or oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

~~RADIOACTIVE EFFLUENTS~~

~~EXPLOSIVE GAS MIXTURE (Hydrogen rich systems not designed to withstand a hydrogen explosion)~~

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N/A

Delete

LIMITING CONDITION FOR OPERATION

3.11.2.5B The concentration of oxygen in the waste gas holdup system shall be limited to less than or equal to 2% by volume whenever the hydrogen concentration exceeds 4% by volume.

APPLICABILITY: At all times.

ACTION:

a. With the concentration of oxygen in the waste gas holdup system greater than 2% by volume but less than or equal 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas holdup system greater than 4% by volume and the hydrogen concentration greater than 2% by volume, immediately suspend all additions of waste gases to the system and reduce the concentration of oxygen to less than or equal to 2% by volume within one hour.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5B The concentrations of hydrogen and oxygen in the waste gas holdup system shall be determined to be within the above limits by continuously monitoring the waste gases in the waste gas holdup system with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

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RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

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LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to ___ curies noble gases (considered as Xe-133).

280,000 per NRC calc method

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank and within 48 hours reduce the tank contents to within the limit.

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

SURVEILLANCE REQUIREMENTS

4.11.2.6 The quantity of radioactive material contained in each gas storage tank shall be determined to be within the above limit at least once per 24 hours when radioactive materials are being added to the tank.

RADIOACTIVE EFFLUENTS

3/4.11.3 SOLID RADIOACTIVE WASTE

DRAFT

LIMITING CONDITION FOR OPERATION

3.11.3 The solid radwaste system shall be OPERABLE and used, as applicable in accordance with a PROCESS CONTROL PROGRAM, for the SOLIDIFICATION and packaging of radioactive wastes to ensure meeting the requirements of 10 CFR Part 20 and of 10 CFR Part 71 prior to shipment of radioactive wastes from the site.

APPLICABILITY: At all times.

ACTION:

a. With the packaging requirements of 10 CFR Part 20 and/or 10 CFR Part 71 not satisfied, suspend shipments of defectively packaged solid radioactive wastes from the site.

b. With the solid radwaste system inoperable for more than 31 days, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status,

   3. A description of the alternative used for SOLIDIFICATION and packaging of radioactive wastes, and

   4. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3.1 The solid radwaste system shall be demonstrated OPERABLE at least once per 92 days by:

a. Operating the solid radwaste system at least once in the previous 92 days in accordance with the PROCESS CONTROL PROGRAM, or

b. Verification of the existence of a valid contract for SOLIDIFICATION to be performed by a contractor in accordance with a PROCESS CONTROL PROGRAM.

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SURVEILLANCE REQUIREMENTS (Continued)

4.11.3.2 THE PROCESS CONTROL PROGRAM shall be used to verify the SOLIDIFICATION of at least one representative test specimen from at least every tenth batch of each type of wet radioactive waste (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions).

a. If any test specimen fails to verify SOLIDIFICATION, the SOLIDIFICATION of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternative SOLIDIFICATION parameters can be determined in accordance with the PROCESS CONTROL PROGRAM, and a subsequent test verifies SOLIDIFICATION. SOLIDIFICATION of the batch may then be resumed using the alternative SOLIDIFICATION parameters determined by the PROCESS CONTROL PROGRAM.

b. If the initial test specimen from a batch of waste fails to verify SOLIDIFICATION, the PROCESS CONTROL PROGRAM shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least 3 consecutive initial test specimens demonstrate SOLIDIFICATION. The PROCESS CONTROL PROGRAM shall be modified as required, as provided in Specification 6.13, to assure SOLIDIFICATION of subsequent batches of waste.

[redacted]

3/4 11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

DRAFT

3.11.4 The dose or dose commitment to any member of the public, due to releases of radioactivity and radiation, from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the total body or any organ (except the thyroid, which shall be limited to less than or equal to 75 mrem) over 12 consecutive months.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2.a, 3.11.1.2.b, 3.11.2.2.a, 3.11.2.2.b, 3.11.2.3.a, or 3.11.2.3.b, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Director, Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, within 30 days, which defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the limits of Specification 3.11.4. This Special Report shall include an analysis which estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources (including all effluent pathways and direct radiation) for a 12 consecutive month period that includes the release(s) covered by this report. If the estimated dose(s) exceeds the limits of Specification 3.11.4, and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR 190 and including the specified information of § 190.11(b). Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. The variance only relates to the limits of 40 CFR 190, and does not apply in any way to the requirements for dose limitation of 10 CFR Part 20, as addressed in other sections of this technical specification.

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

SURVEILLANCE REQUIREMENTS

4.11.4 Dose Calculations Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Specifications 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the ODCM.

[redacted]

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING DRAFT

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission, in the Annual Radiological Operating Report, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the level of radioactivity in an environmental sampling medium exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days from the end of the affected calendar quarter a Report pursuant to Specification 6.9.1.13. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{limit level (1)}} + \frac{\text{concentration (2)}}{\text{limit level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to an individual is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or fresh leafy vegetable samples unavailable from one or more of the sample locations required by Table 3.12-1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause of the unavailability of samples and identifies locations for obtaining replacement samples. The locations from which samples were unavailable may then be deleted from those required by Table 3.12-1, provided the locations from which the replacement samples were obtained are added to the environmental monitoring program as replacement locations.

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

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SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the locations given in the table and figure in the ODCM and shall be analyzed pursuant to the requirements of Tables 3.12-1 and 4.12-1.

TABLE ~~18.1~~ 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
1. AIRBORNE			
a. Radioiodine and Particulates	(1)	Continuous 7 day samples	Gross beta at least 72 hrs. after collection weekly. Gamma spectrum analysis if above $1pCi/m^3$. Filters composited quarterly for gross alpha, gamma spectrum and radiostrontium analyses. Charcoal filters analyzed for I-131 within 8 days of collection by gamma spectrum analysis.
2. DIRECT RADIATION	(1)	Four $CaSO_4$ (Dy) TLD's at each station collected quarterly. Two LiF TLD's at each station collected annually.	Gamma dose quarterly
3. WATERBORNE			
a. Ocean Water	(2)	Bimonthly	Gross beta bimonthly. Gamma spectrum for Cs-137 if gross beta greater than 30 pCi/l. Radiostrontium analysis if Cs-137 is present. Tritium semiannually on composited samples.

(1) See Figure 3.18.1

(2) See Figure 3.18.2

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TABLE [redacted] (continued)
3.17.1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
b. Drinking Water	(1)	Samples collected monthly at each station.	Gross beta and gross alpha monthly on filtrate and solids. Tritium on filtrate. Gamma spectrum analysis for Cs-137 and I-131 if gross beta above 30 pCi/l. Radiostrontium analysis if Cs-137 above 200 pCi/l. Composited quarterly for gross beta and gross alpha on filtrate and solids and tritium on filtrate.
c. Beach Sand	(2)	Semiannually.	Gamma spectrum analysis semiannually.
d. Ocean Bottom Sediments	(2)	Semiannually.	Gamma spectrum analysis semiannually.
4. INGESTION			
a. Nonmigratory marine animals	(2)	Semiannually. Each sample will include: 1. Fish - 2 adult species such as perch and sheepshead.	Gamma and tritium analyses of each species flesh quarterly. If Cs-137 levels above 6nCi/Kg radiostrontium analysis will be performed.

(1) See Figure 3.18.1

(2) See Figure 3.18.2

TABLE ~~3.18.1~~ 3.12-1 (continued)

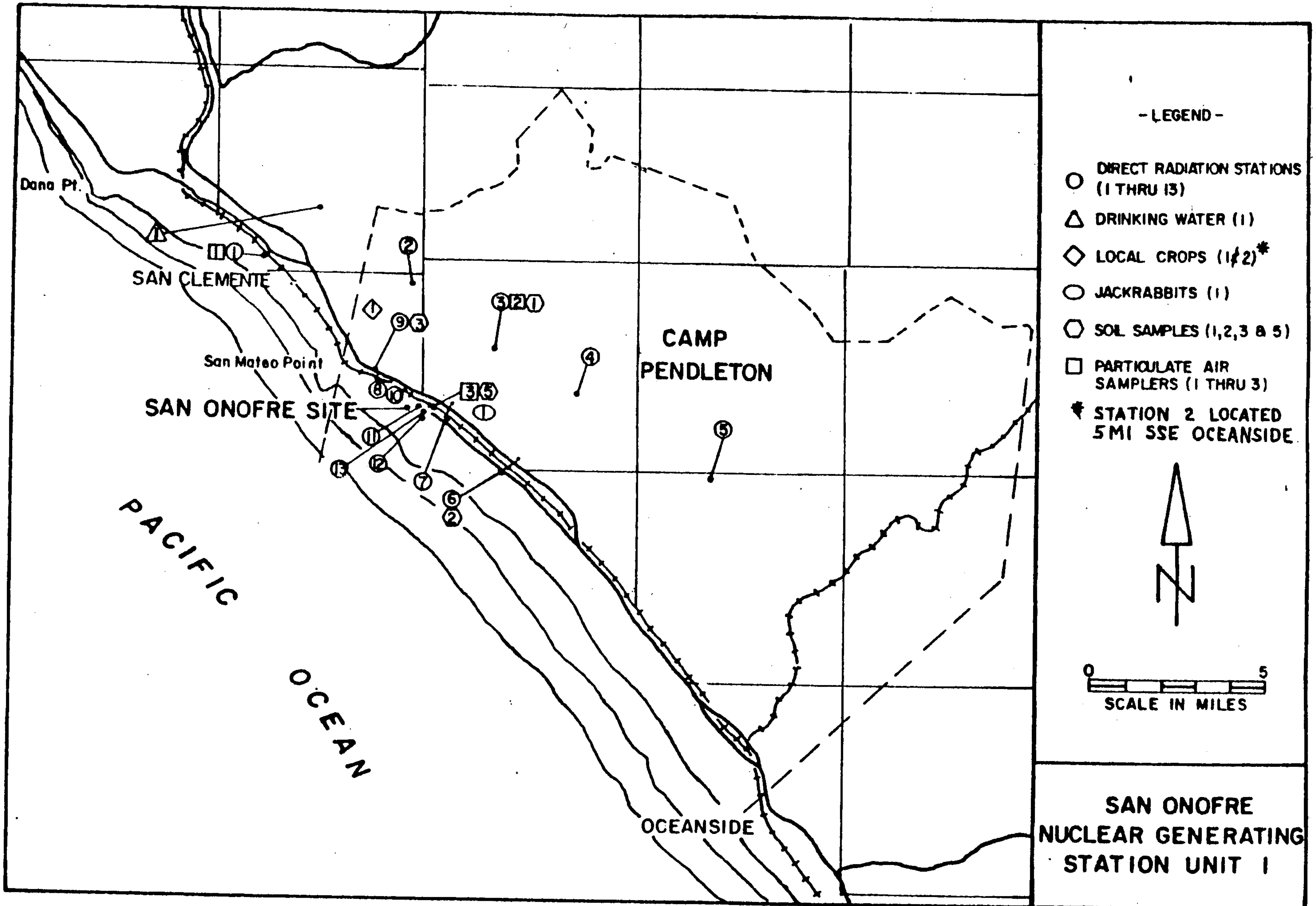
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway and/or Sample	Sample Locations	Sampling and Collection Frequency	Type and Frequency of Analysis
		2. Crustaceae - such as crab or lobster	
		3. Mollusks - such as seahares	
b. Local Crops	(1)	Representative vegetables normally 1 leafy and 1 fleshy collected at harvest time. At least 2 vegetables semi-annually from each station.	Gamma, tritium and radiostrontium analyses twice per year.
c. Kelp	(2)	Semiannually.	Gamma and tritium analysis semiannually.
5. OTHER			
a. Soil Sampling	(1)	Annually	Gamma and radiostrontium analyses annually.
b. Jack Rabbit Sampling	(1)	Semiannually.	Thyroid analyzed for I-131; Femur analyzed for Sr-90 and 89 levels; and gamma spectrum analysis of flesh semiannually.

(1) See Figure ~~3.18.1~~ 3.12-1

(2) See Figure ~~3.18.2~~ 3.12-2

(3) See Figure 3.12-3



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-26d-

FIGURE ~~3.18.1~~ 3.12-1 TERRESTRIAL MONITORING FOR THE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

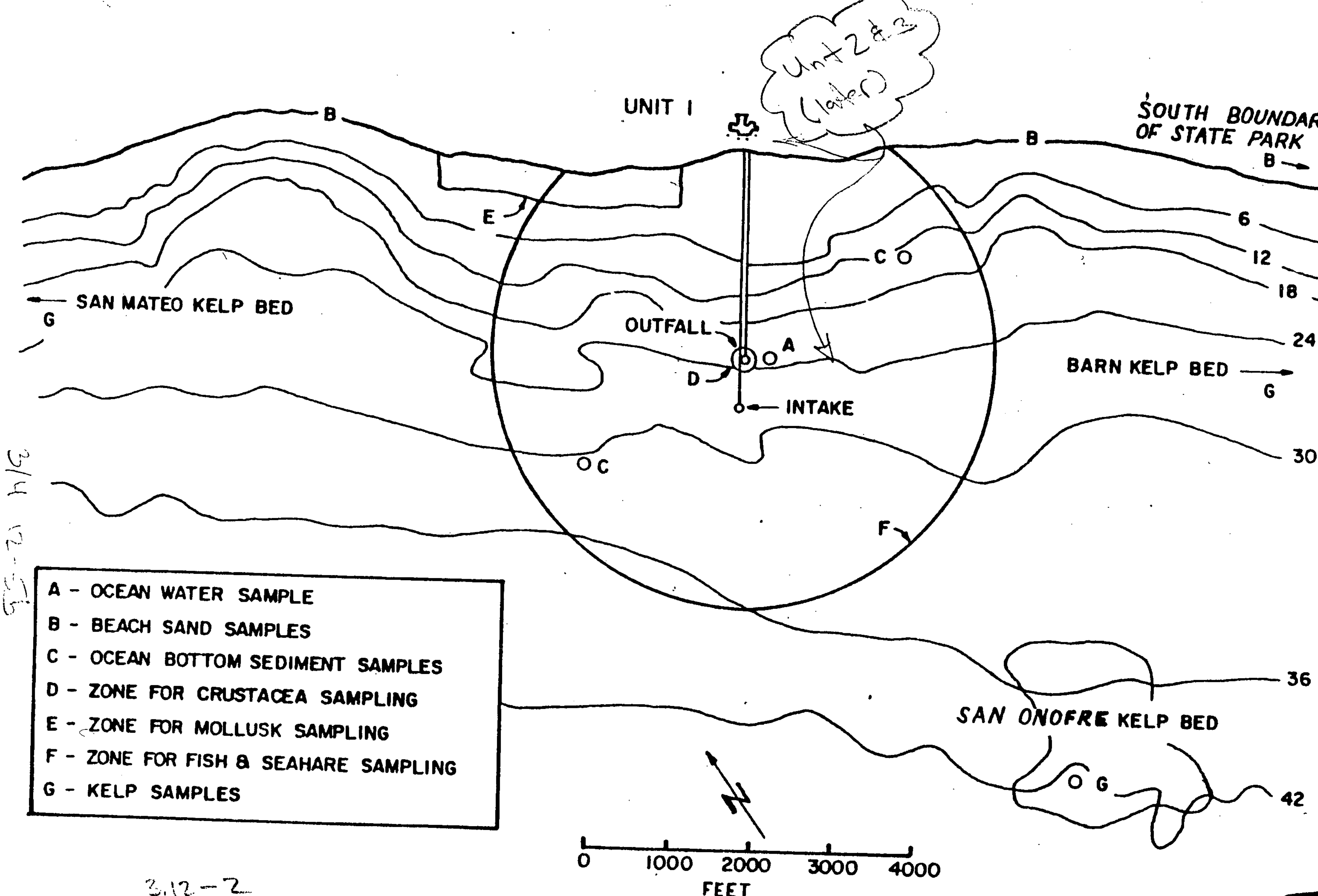

FIGURE 3.12-2 OCEANOGRAPHIC MONITORING STATIONS FOR THE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

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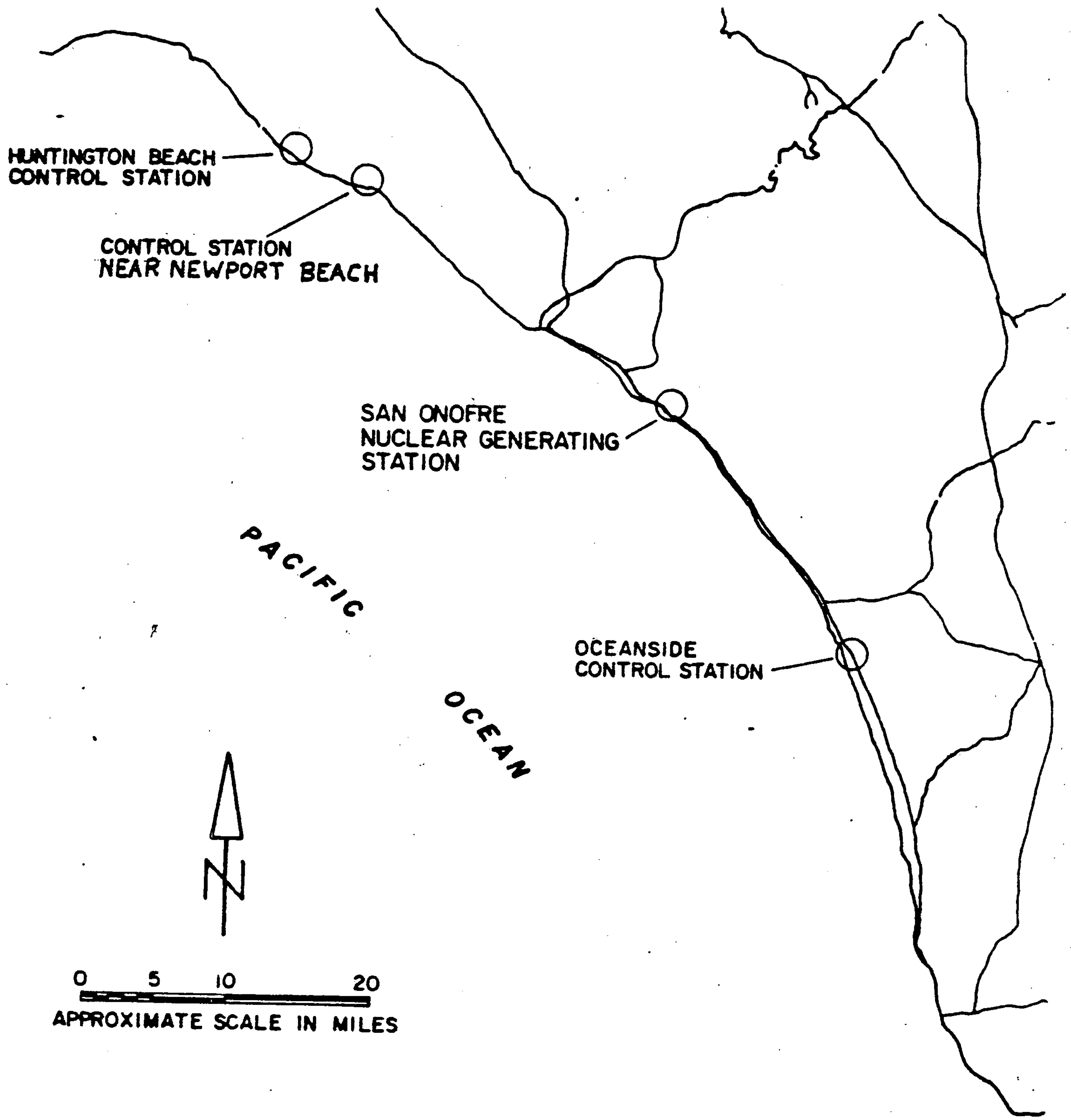

FIGURE ~~3.18.3~~ 3.12-3 CONTROL STATIONS FOR THE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

3/4 12-5c

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/l)	Airborne Particulate or Gases (pCi/m^3)	Marine Animals ~~Fish~~ (pCi/Kg, wet)	~~Milk (pCi/l)~~	Local Crops ~~Food Products~~ (pCi/Kg, wet)
H-3	3 ~~2~~ $\times 10^4$ (a)				
~~Mn-54~~	~~1×10^3~~		~~3×10^4~~		
~~Fe-59~~	~~4×10^2~~		~~1×10^4~~		
Co-58	1×10^3		3×10^4		
Co-60	3×10^2		1×10^4		
~~Zn-65~~	~~3×10^2~~		~~2×10^4~~		
~~Zr-Nb-95~~	~~4×10^2~~				
I-131	2	0.9		~~3~~	1×10^2
~~Cs-134~~	~~30~~	~~10~~	~~1×10^3~~	~~60~~	~~1×10^3~~
Cs-137	50	20	2×10^3	~~70~~	2×10^3
~~Ba-La-140~~	~~2×10^2~~			~~3×10^2~~	

(a) For drinking water samples. This is 40 CFR Part 141 value.

4.12-1

TABLE [redacted]

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)

Analysis	Drinking Water (pCi/l)	Ocean Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Marine Animals (pCi/g)[2]	Local Crops (pCi/g)[2]	Kelp (pCi/g)[2]	Beach Sand (pCi/g)	Ocean Bottom (pCi/g)	Jack Rabbit (pCi/g)
Gross beta	0.5 (0.1)[1]	0.5	.003						
Gross alpha	5.0 (0.3)[1]	5.0	.003						
H-3	200			11	11	10			
Sr-90	1		.001		.04				2.0
Sr-89									3.0
Cs-137	5	6		0.03	.03	.06	.07	.07	.06
I-131	10		.04		.03	.2			.05 (5.0)[3]
Co-58				0.1		.13		.07	
Co-60				0.2		.20		.11	
Ag-110m				0.4		.07		.07	

Footnotes

1. Solids
2. Dry wt.
3. Thyroid

TABLE 4.12-1 (Continued)

TABLE NOTATION

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a. The LLD is the smallest concentration of radioactive material in a sample that will be detected with 95% probability with 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system (which may include radiochemical separation):

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

- LLD is the "a priori" lower limit of detection as defined above (as picocurie per unit mass or volume),

  s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (as counts per minute),

  E is the counting efficiency (as counts per transformation),

  V is the sample size (in units of mass or volume),

  2.22 is the number of transformation per minute per picocurie,

  Y is the fractional radiochemical yield (when applicable),

  λ is the radioactive decay constant for the particular radionuclide, and

  Δt is the elapsed time between sample collection (or end of the sample collection period) and time of counting (for environmental samples, not plant effluent samples).

The value of s_b used in the calculation of the LLD for a detection system shall be based on the actual observed variance of the background counting rate or of the counting rate of the blank samples (as appropriate) rather than on an unverified theoretically predicted variance. In calculating the LLD for a radionuclide determined by gamma-ray spectrometry, the background shall include the typical contributions of other radionuclides normally present in the samples (e.g., potassium-40 in milk samples). Typical values of E, V, Y and Δt shall be used in the calculations.

TABLE 4.12-1 (Continued)

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TABLE NOTATION

b. LLD for drinking water.

c. Other peaks which are measurable and identifiable, together with the radionuclides in Table 4.12-1, shall be identified and reported.

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3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify the location of the nearest milk animal, the nearest residence and the nearest garden* of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of five miles. (For elevated releases as defined in Regulatory Guide 1.111, Revision 1, July 1977, the land use census shall also identify the locations of all milk animals and all gardens of greater than 500 square feet producing fresh leafy vegetables in each of the 16 meteorological sectors within a distance of three miles.)

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) which yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of any other report required by Specification 6.9.1., prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location(s).

b. With a land use census identifying a location(s) which yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the new location. The new location shall be added to the radiological environmental monitoring program within 30 days. The sampling location, excluding the control station location, having the lowest calculated dose or dose commitment (via the same exposure pathway) may be deleted from this monitoring program after (October 31) of the year in which this land use census was conducted.

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted at least once per 12 months between the dates of (June 1 and October 1) using that information which will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities.

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*Broad leaf vegetation sampling may be performed at the site boundary in the direction sector with the highest D/Q in lieu of the garden census.

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RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program which has been approved by the Commission.

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions taken to prevent a recurrence to the Commission in the Annual Radiological Environmental Operating Report.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program and in accordance with ~~the ODCM (or participants in the EPA crosscheck program shall provide~~ the EPA program code designation ~~for the unit)~~ shall be included in the Annual Radiological Environmental Operating Report.

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BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

[redacted]

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NOTE

The ~~summary statements contained in this section~~ BASES in the succeeding pages summarize the reasons for ~~provide the bases of~~ the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not considered a part of these Technical Specifications.

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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 measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For 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, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this 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.

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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, action 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

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3.0.5 (Continued)

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, action 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.

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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.

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 5.15% 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 2% delta k/k shutdown margin provides adequate protection.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the 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.

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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 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 2.0% 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 ~~of~~ or 53,500 gallons of 1720 ppm borated water from the refueling water tank. However, for the purpose of consistency the minimum required volume of water in Specification 3.1.2.8 is identical to the more restrictive value of Specification 3.5.4.

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 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5465 gallons of 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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not consistent w/ BAMU req in 3.1.2.7; CE to provide correct #

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 7.4 and 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.

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BORATION SYSTEMS (Continued)

The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics.

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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.

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 520°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.

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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.

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 520°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.

Insert "A"

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INSERT "A"

Several design steps were employed to accommodate the possible CEA guide tube wear which could arise from CEA vibrations when fully withdrawn. One design item was to define the drop limit for four (4) CEAs (#88-91) to be at 85% inserted, as opposed to the 90% insertion position. This change results from a decrease of 9.75 inches in the poison section of these four CEAs. For 90% of these CEA's poison to be inserted, the CEA only need to be inserted 85% of the core's length. The second step taken to accommodate the guide tube wear concern was to redefine the fully withdrawn position, in some cases, to be within 3.0 inches of the full out position. This redefinition allows a programmed insertion schedule to be employed. The programmed insertion schedule specified the CEAs to be cycled between the full out position and 3.0 inches inserted over the fuel cycle. This cycling will distribute the possible guide tube wear over a larger area, thus minimizing the effects of any CEA guide tube wear. Therefore, regulating CEAs are considered to be fully withdrawn in accordance with Figure 3.1-2 when withdrawn to within 3.0 inches of the full out position.

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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.

Duplicate Sentence

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.

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3/4.2.1 LINEAR HEAT RATE

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 (1.053) includes an Fxy F 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, and thermal power measurement.

CE to verify

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.

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3/4.2.2 RADIAL PEAKING FACTORS

Limiting the values of the planar radial peaking factors (F^C_{xy}) used in the COLSS and CPCs to values equal to or greater than the measured planar radial peaking factors (F^m_{xy}) 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

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

AZIMUTHAL POWER TILT - T_q (Continued)

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

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~~ 1.19 would 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_{xy} measurement uncertainty factor of ~~1.080~~ (1.053), 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, AND THERMAL POWER measurement

CE to verify

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.

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3/4.2.5 RCS FLOW RATE

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

This specification is provided to ensure that the assumptions used for the initial conditions of the LOCA safety analyses remain valid.

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3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE ~~AND~~ and ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

The OPERABILITY of the Reactor protective and Engineered Safety Features Actuation System instrumentation and bypasses ensure that 1) the associated Engineered Safety Features Actuation System action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the reactor ~~trip~~ protective and ESF actuation associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING ALARM INSTRUMENTATION

The OPERABILITY of the radiation monitoring alarm 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.

ESFAS requirements on such channels now in ESFAS tables

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.

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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 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations".

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 RADIOACTIVE LIQUID EFFLUENT INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.9 RADIOACTIVE GASEOUS EFFLUENT INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the procedures in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring (and controlling) the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 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.

[redacted]

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

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 MODE 4, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two trains be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling trains to be OPERABLE.

In MODE 5, single failure considerations require that two shutdown cooling loops 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) restricitng 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 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 x 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 provides a diverse means of protection against RCS overpressurization at low temperatures.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

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.

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In MODE 4, a single reactor coolant loop or shutdown cooling train provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two trains be OPERABLE. Thus, if the reactor coolant loops are not OPERABLE, this specification requires two shutdown cooling trains to be OPERABLE.

In MODE 5, single failure considerations require that two shutdown cooling loops be OPERABLE.

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(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) restricitng 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 SAFETY VALVES

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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 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.

N/A SO2

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.

3/4.4.3 PRESSURIZER

The limit on the maximum water volume in the pressurizer assures that the parameter is maintained within the normal steady-state envelope of operation assumed in the SAR. 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 enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

no PORV's at SO2/3

3/4.4.4 RELIEF VALVES

The power operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs 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.~~5~~ 4 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.

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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) 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) 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)% of the tube nominal wall thickness.~~ Figure 4.4-1. 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~~ 6.8.1 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.~~6~~5 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.~~6~~5.1 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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LEAKAGE DETECTION SYSTEMS (Continued)

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.~~6~~5.2 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.~~

SO 2/3 RCP's do not use seal inject

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) 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.~~7~~6 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

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BASES

LEAKAGE DETECTION SYSTEMS (Continued)

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.~~6~~5.2 OPERATIONAL LEAKAGE

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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.~~8~~7 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 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 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 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.

3/4.4.9 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)

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) and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." 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 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. 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 - @ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction-ft lb

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Figure B3/4.4-1

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PRESSURE/TEMPERATURE LIMITS (Continued)

(later)

~~The OPERABILITY of the Shutdown Cooling System relief valve 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. The Shutdown Cooling System relief valve 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.~~10~~ 9 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.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Reactor Coolant System (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. Note!

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.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 350°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

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.

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 flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 REFUELING WATER STORAGE TANK (RWST)

The OPERABILITY of the RWST as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWST minimum volume and boron concentration ensure that 1) sufficient water is available within containment to permit recirculation cooling flow to the core, and 2) the reactor will remain subcritical in the cold condition following mixing of the RWST and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

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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 8.0 and 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.

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.~~5~~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 5.0 psig and 2) the containment peak pressure does not exceed the design pressure of 60 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 55.7 psig. The limit of 1.5 psig for initial positive containment pressure will limit the total pressure to 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 steam line break accident.

3/4.6.1.~~7~~6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 55.7 psig in the event of a steam line break accident. The measurement of containment tendon lift off force, the tensile tests of the tendon wires 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.

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3/4.6.1.~~8~~7 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves are required to be closed with power to the valves removed 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.

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.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the iodine removal system ensures that sufficient NaOH is added to the containment spray in the event of a LOCA. The limits on NaOH volume and concentration ensure a pH value of between 8.0 and 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 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

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.

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.

3/4.6.3 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.

3/4.6.4 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 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 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.

3/4.7.1 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 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, 1974 Edition. The total relieving capacity for all valves on all of the steam lines is 15,473,628 lbs/hr which is 102.3 percent of the total secondary steam flow of 15,130,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of 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:

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)

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SAFETY VALVES (Continued)

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)

3/4.7.1.2 Auxiliary FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss of off-site power.

Each electric driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 700 gpm at a pressure of 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 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 TANKS

The OPERABILITY of the condensate storage tanks 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 of 70°F and 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 SALT WATER COOLING SYSTEM

The OPERABILITY of the salt water cooling 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.

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3/4.7.5 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 ~~exposure~~ to personnel occupying the control room to 5 rem or less whole body, or exposure 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.

~~3/4.7.6 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.)~~

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STANDBY NUCLEAR SERVICE WATER POND (Continued)

exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants," March 1974.

3/4.7.6 CONTROL AREA VENTILATION SYSTEM

The OPERABILITY of the control area ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A of 10 CFR 50. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.7 AUXILIARY BUILDING FILTERED EXHAUST SYSTEM (ECCS PUMP ROOM EXHAUST AIR FILTRATION SYSTEM)

The OPERABILITY of the auxiliary building filtered exhaust system ensures that radioactive materials leaking from the ECCS equipment within the auxiliary building 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. ANSI N510-1975 will be used as a procedural guide for surveillance testing.

3/4.7.~~8~~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 non-safety 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.

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SNUBBERS (Continued)

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 adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18-month intervals. The sample size will be determined in accordance with Figure 4.7.1 which has been developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan. Hydraulic snubbers and mechanical snubbers may each be treated as a different entity for the above surveillance programs.

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.

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SNUBBERS (Continued)

~~The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life. The requirements for the maintenance of records and the snubber service life review are not intended to affect plant operation.~~

3/4.7.~~8~~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.~~9~~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, and fire hose stations. 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.

E

FIRE SUPPRESSION SYSTEMS (Continued)

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant. The requirement for a twenty-four hour report to the Commission provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

3/4.7.~~10~~ 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.

~~3/4.7.11 AREA TEMPERATURE MONITORING~~

~~The area temperature limitations ensure that safety related equipment will not be subjected to temperatures in excess of their environmental qualification temperatures. Exposure to excessive temperatures may degrade equipment and can cause a loss of its OPERABILITY. The temperature limits include an allowance for instrument error of ( )°F.~~

N/A to SO2/3

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, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

~~The Surveillance Requirements 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 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.

DRAFT

Attachment E

p1/2

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 Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9, "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, 1.108, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel-Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage onfloat charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

DRAFT

Attachment
p2/2

A.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.3 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.

The surveillance requirements applicable to lower voltage circuit breakers and fuses provide assurance of breaker and fuse reliability by testing at least one representative sample of each manufacturer's brand of circuit breaker and/or fuse. Each manufacturer's molded case and metal case circuit breakers and/or fuses are grouped into representative samples which are then tested on a rotating basis to ensure that all breakers and/or fuses are tested. If a wide variety exists within any manufacturer's brand of circuit breakers and/or fuses, it is necessary to divide that manufacturer's breakers and/or fuses into groups and treat each group as a separate type of breaker or fuses for surveillance purposes.

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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.

The thermal overload protection contact integral with the motor starter of each valve listed in Table 3.8.2 is permanently bypassed.

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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.

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3/4.9.6 REFUELING MACHINE

The OPERABILITY requirements for the refueling machine ensure that: 1) refueling machine will be used for movement of CEAs and fuel assemblies, 2) each 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.

N/A to 50243

~~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.

DRAFT

BASES

3/4.9.~~10~~ 9 and 3/4.9.~~11~~ 10 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 ~~STORAGE VENTILATION UNIT~~ 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 automatic for at least 10 hours over a 31 day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

DRAFT

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.

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR 50, to an individual, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." ~~Also, for fresh water sites with drinking water supplies which can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations in the finished drinking water that are in excess of the requirements of 40 CFR 141.~~ The dose calculations in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

N/A

~~This specification applies to the release of liquid effluents from each reactor at the site.~~ For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID WASTE TREATMENT

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

~~3/4.11.1.4 LIQUID HOLDUP TANKS~~

~~Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10 CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area.~~

N/A

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 for unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of an individual in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For individuals who may at times be within the site boundary, the occupancy of the individual will be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to an individual at or beyond the site boundary to less than or equal to 500 mrem/year to the total body or to less than or equal to 3000 mrem/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to an infant via the cow-milk-infant pathway to less than or equal to 1500 mrem/year for the nearest cow to the plant.

DRAFT

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents will be kept "as low as is reasonably achievable". The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The dose calculations established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at the site boundary are based upon the historical average atmospheric conditions.

3/4.11.2.3 DOSE - RADIOIODINES, RADIOACTIVE MATERIALS IN PARTICULATE FORM AND RADIONUCLIDES OTHER THAN NOBLE GASES

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of an individual through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methods for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109,

"Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for radioiodines, radioactive materials in particulate form and radionuclides other than noble gases are dependent on the existing radionuclide pathways to man, in the unrestricted area. The pathways which were examined in the development of these calculations were: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, 3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and 4) deposition on the ground with subsequent exposure of man.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure".

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR 190. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant radioactive effluents exceed twice the design objective doses of Appendix I. For sites containing up to 4 reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within the reporting requirement level. The Special Report will describe a course of action which should result in the limitation of dose to a member of the public for 12 consecutive months to within the 40 CFR 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 5 miles must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. An individual is not considered a member of the public during any period in which he/she is engaged in carring out any operation which is part of the nuclear fuel cycle.

BASES

3/4.12.1 MONITORING PROGRAM

The radiological monitoring program required by this specification provides measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides, which lead to the highest potential radiation exposures of individuals resulting from the station operation. This monitoring program thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first three years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The detection capabilities required by Table 4.12-1 are state-of-the-art for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an a priori (before the fact) limit representing the capability of a measurement system and not as a posteriori (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidably small sample sizes, the presence of interferring nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors will be identified and described in the Annual Radiological Environmental Operating Report.

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of unrestricted areas are identified and that modifications to the monitoring program are made if required by the results of this census. The best survey information from the door-to-door, aerial or consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 500 square feet provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/year) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were used, 1) that 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/square meter.

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid.

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SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area shall be as shown in Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone shall be as shown in Figure 5.1-2.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

5.1.3 The site boundary for gaseous effluents shall be as shown in Figure 5.1-3.

SITE BOUNDARY FOR LIQUID EFFLUENTS

5.1.4 The site boundary for liquid effluents shall be as shown in Figure 5.1-4.

5.2 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.

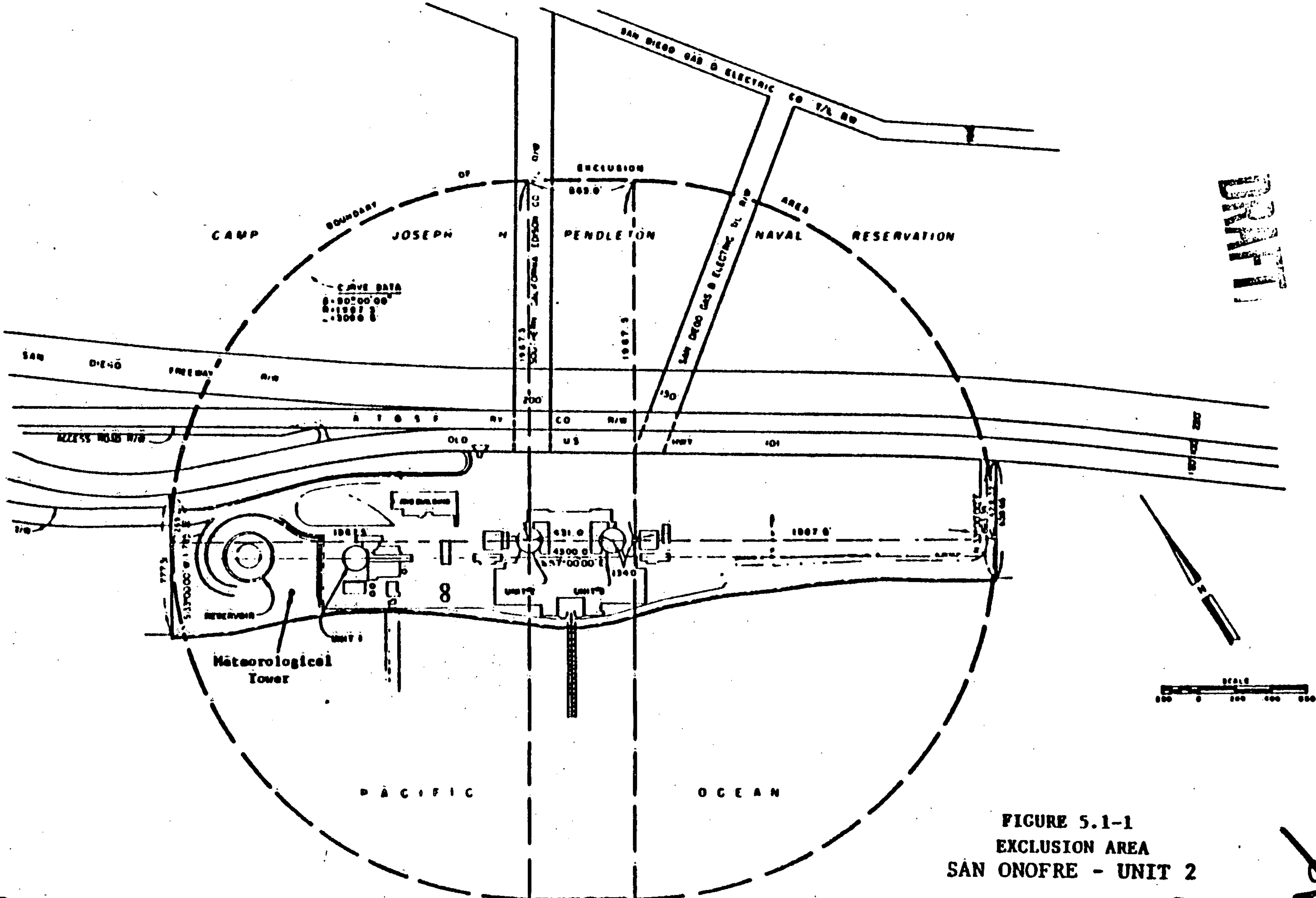
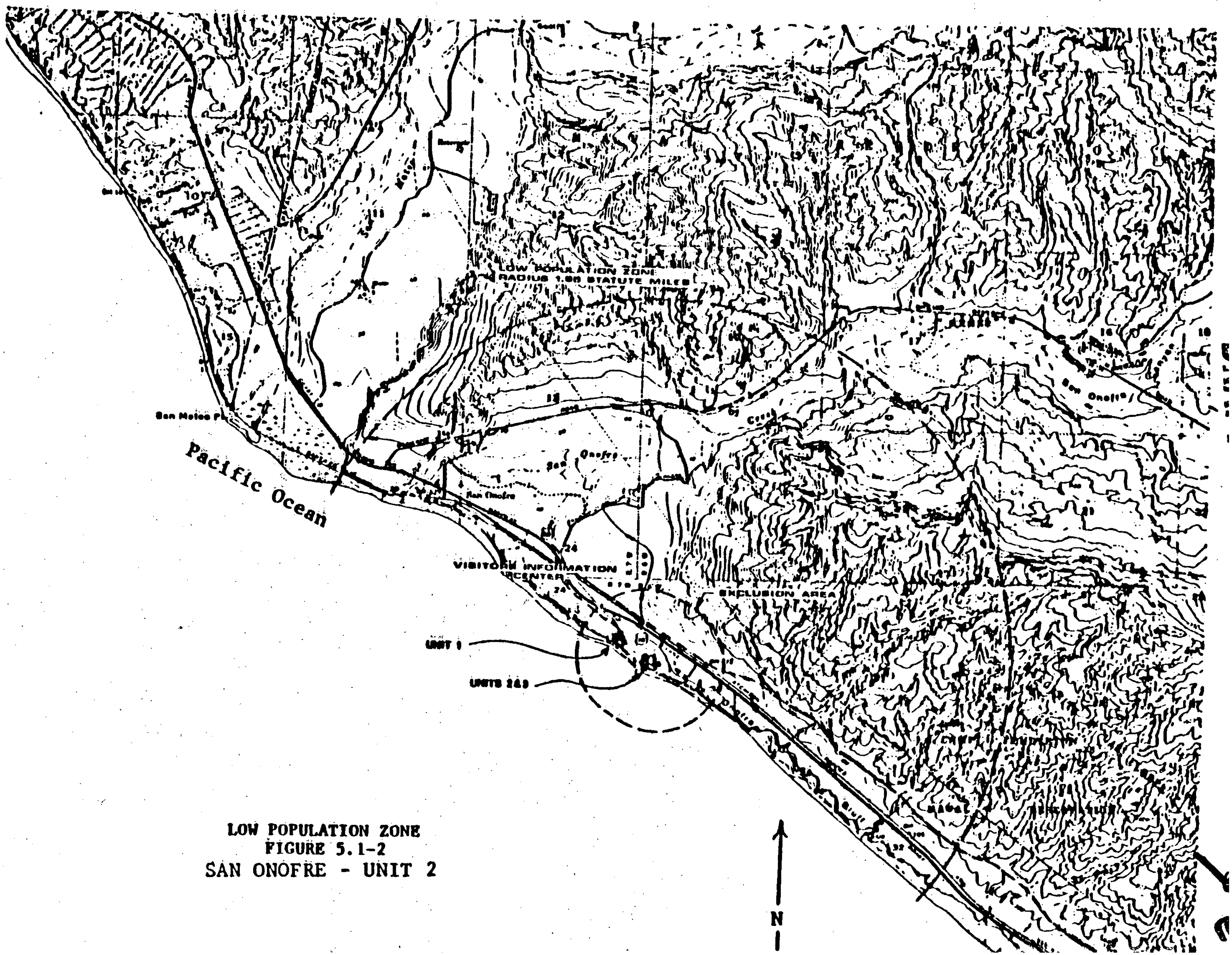


FIGURE 5.1-1
EXCLUSION AREA
SAN ONOFRE - UNIT 2



LOW POPULATION ZONE
FIGURE 5.1-2
SAN ONOFRE - UNIT 2

This figure shall consist of a map of the site area showing the perimeter of the site and locating the points where gaseous effluents are released. If on-site land areas subject to radioactive materials in gaseous waste are utilized by the public for recreational or other purposes, then these areas shall be identified by occupancy factors and the licensee's method of occupancy control. The figure shall be sufficiently detailed to allow identification of structures and release point elevations, and areas within the site boundary that are accessible by members of the general public. See NUREG-0133 for additional guidance.

SITE BOUNDARY FOR GASEOUS EFFLUENTS

FIGURE 5.1-3

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This figure shall consist of a map of the site area showing the perimeter of the site and locating the points where liquid effluent leaves the site. If on-site water areas containing radioactive wastes are utilized by the public for recreational or other purposes, the points of release to these water areas shall be identified. The figure shall be sufficiently detailed to allow identification of structures near the release points and areas within the site boundary where ground and surface water is accessible by members of the general public. See NUREG-0133 for additional guidance.

SITE BOUNDARY FOR LIQUID EFFLUENTS

FIGURE 5.1-4

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5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 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.

CE to provide

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.

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 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 0.014 delta k/k for uncertainties as described in Section 9.1 of the FSAR.

b. A nominal 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 61'6".

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 (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.~~
Per FSAR Table 3.9-1	~~5~~ 2 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 and 1000 pressurizer spray cycles total with pressurizer/spray water $\Delta T > 200^{\circ}F$ 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 $\Delta T > 200^{\circ}F$.

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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.

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SECTION 6.0

ADMINISTRATIVE CONTROLS

[redacted]

Change Notes (not part of spec)

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June 3, 1981

Explanation for changes to SONGS 2&3 Administrative Controls Technical Specifications

Comment No.	STS Section	Explanation
1	6.1.2	A management direction signed by a recognized station approved authority, which is in effect at all times permanently establishes the Watch Engineer's authority, where as a corporate memorandum to all personnel is generally informational in nature and must be reiterated in a formally approved procedure to be enforced.
2.	6.2.2.b Table 6.2-1	The office for Units 2&3 SRO's is located near, but not within the boundaries of, the Control Room. This area will be described in Station procedures.
3	6.2.2.6	Absence of a health physics technician or fire brigade member could be construed to require a plant shutdown under the present wording. However, a shutdown is not technically justified.
4	Table 6.2-1	This change is made for clarification. Operating Instructions will specify a chain of command such that the senior man present is designated the shift supervisor/Watch Engineer.
5	6.2.3.2 6.5.2	Changes made to conform to FSAR Chapter 13 and previous NRC agreements.
6	6.31	This ANSI standard was the requirement of all previous NRC questions and the TQAM.
7	6.5.1.7.a,c 6.5.2.7.a	Change made to conform to revised Specifications 6.8.2 and 6.8.3.
8		Deleted

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Comment No.	SIS Section	Explanation
9	6.5.2.7.j 6.5.28	All audits are reassigned to the QA organization and reviewed by the Safety Group as described in FSAR Chapter 13.
10	6.5.2.10	Change made to reflect that independent reviews are performed by a staff organization rather than a standing committee.
11	6.5.3	Nuclear Control Board organization is updated to describe their proposed activities.
12	6.6	This section removed since it is entirely redundant to 6.9.1.7, 6.5.1.6., and 6.5.2.7
13	6.7.1.a	This action was corrected since Specification 2.0 requires more than going to hot standby in some cases.
14	6.8.1	The applicable revision to R. G. 1.33 was changed to agree with the TQAM.
15	6.8.2 6.8.3	These sections were revised to resolve SCE and NRC Region V concerns on the workload of the OSRC. Procedure changes undergo independent technical reviews by appropriate members of the OSRC. An OSRC meeting is not required. This method meets the requirements of 10 CFR 50.59 and recent ANS 3.1 draft revisions.
16	6.10.1.g 6.10.1.h 6.10.2.L 6.10.2.m	These requirements are covered under 6.10.1.d.
17	6.10.1.e 6.10.2.j	Changed to agree with 10 CFR 50.59.
18	6.10.1.i 6.10.3	This change made to conform to R. G. 1.88 which endorses ANSI N45.2.9-1974.
19	6.3.1.6.k	The intent of this item is already covered by OSRC responsibility e and prompt reporting requirement 6.9.1.8.j.
20	6.5.1.6.L	This item was revised to agree with Specification 6.15.

6.1 RESPONSIBILITY

6.1.1 The ~~Plant~~ Station Manager shall be responsible for overall unit operation and shall delegate in writing the succession to this responsibility during his absence.

6.1.2 The Watch Engineer (or during his absence from the Control Room, a designated individual) shall be responsible for the Control Room command function. A management directive to this effect, signed by the ~~Vice President~~ Manager of Nuclear Operations shall be ~~reissued to all station personnel on an annual basis~~ in effect at all times. Note 1

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The Unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the Control Room when fuel is in the reactor. In addition, while the unit is in MODE 1, 2, 3 or 4, at least one licensed Senior Reactor Operator shall be in the Control ~~Room~~ Room area. Note 2

c. A chemical-radiation protection technician# shall be on site when fuel is in the reactor.

d. All CORE ALTERATIONS shall be observed and directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

e. A site Fire Brigade of at least 5 members shall be maintained onsite at all times.# The Fire Brigade shall not include (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 ~~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.

Note 3

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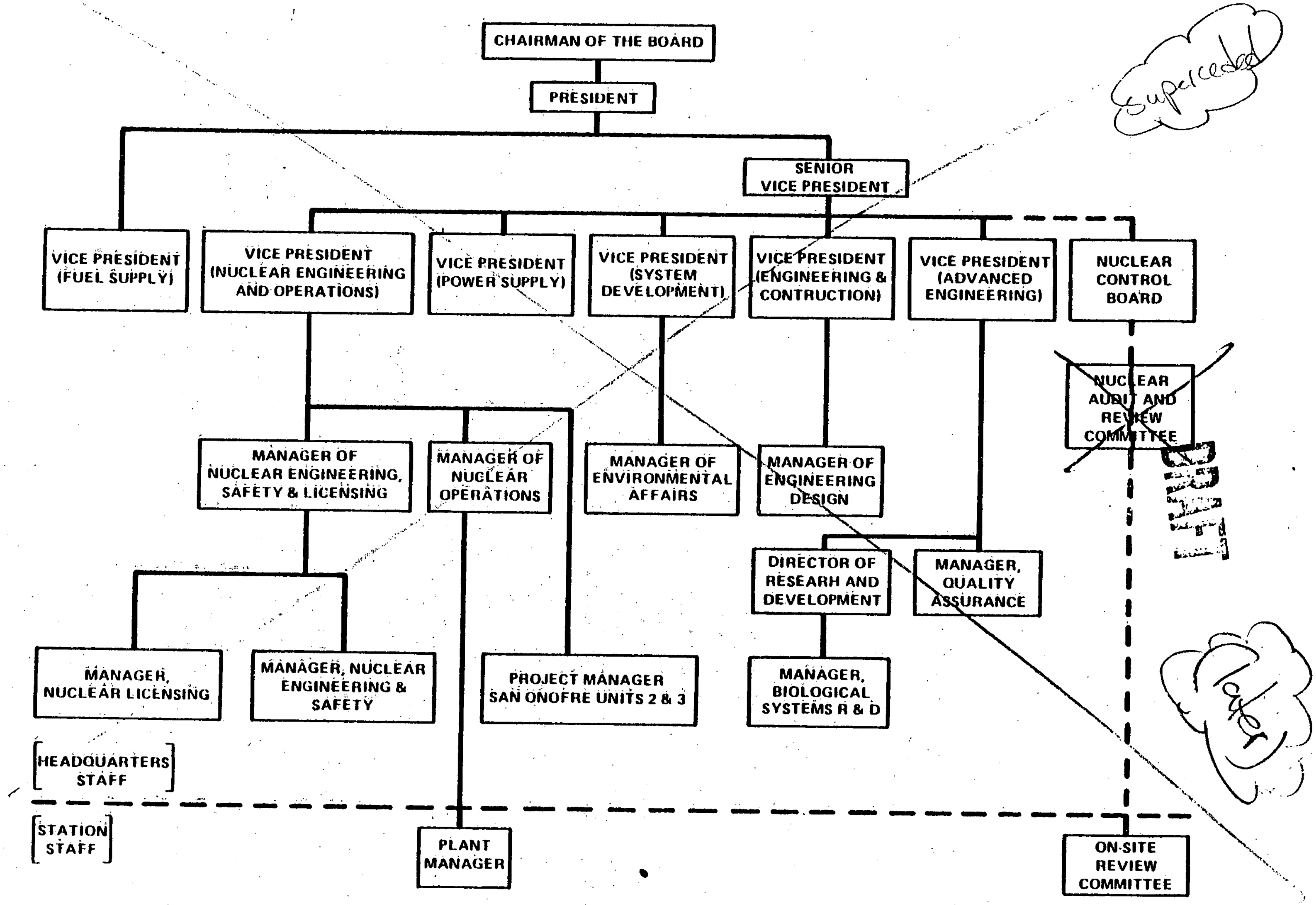

Figure 6.2-1
OFFSITE ORGANIZATION
SAN ONOFRE NUCLEAR GENERATING STATION -- UNIT 2

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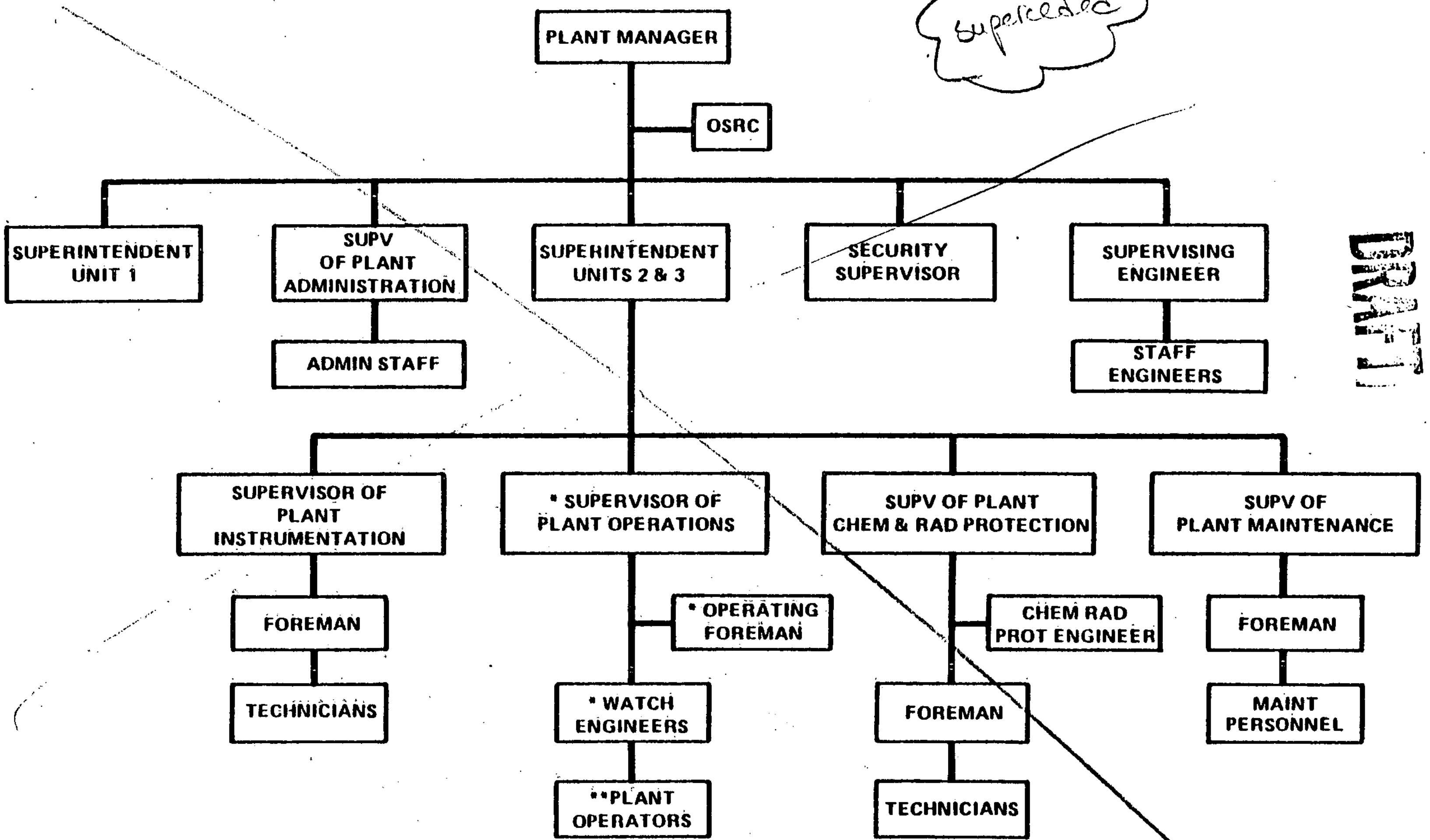

* SENIOR REACTOR OPERATOR LICENSE REQUIRE

** CONTROL AND ASSISTANT CONTROL OPERATORS ARE HOLDERS OF REACTOR OPERATOR LICENSES

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Figure 6.2-2
UNIT ORGANIZATION
SAN ONOFRE NUCLEAR GENERATING STATION – UNIT 2

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Table 6.2-1

MINIMUM SHIFT CREW COMPOSITION

POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODES 1, 2, 3 & 4	MODES 5 & 6
WE	1	1
SRO	1	None
RO	2	1
AO	2	1
STA	1	None

WE - Watch Engineer with a Senior Reactor Operators License on Unit 2
SRO - Individual with a Senior Reactor Operators License on Unit 2
RO - Individual with a Reactor Operators License on Unit 2
AO - Auxiliary Operator
STA - Shift Technical Advisor

Note 4

~~Except for the Watch Engineer,~~ The Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

Note 2

During any absence of the Watch Engineer from the Control Room area while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Watch Engineer 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.

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6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP ~~(ISEG)~~

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of off-duty Shift Technical Advisors ~~at least five, dedicated, full time~~ engineers located on site.

Note 5

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical.

AUTHORITY

6.2.3.4 The ISEG shall make detailed recommendations for revised procedures, equipment modifications, or other means of improving plant safety to ~~(a high level corporate official who is not in the management chain for power production).~~ the Supervisor, Nuclear Safety Group.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall serve in an advisory capacity to the Watch Engineer on matters pertaining to the engineering aspects assuring safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

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Note 6

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ~~(an~~ ANSI Standard ~~agreed to by the NRC staff)~~ for comparable positions and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, except for the ~~(Radiation Protection~~ Health Physics Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

experience

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*Not responsible for sign-off function.

6.4 TRAINING

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Manager, Nuclear Training

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the ~~Plant Manager~~ Manager, Nuclear Training and shall meet or exceed the requirements and recommendations of Section (5.5) of ~~(an~~ ANSI Standard N18.1-19 ~~Standard agreed to by the NRC staff)~~ and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Section A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience identified by the ISEG.

Note 6

6.5 REVIEW AND AUDIT

6.5.1 ONSITE REVIEW COMMITTEE (OSRC)

FUNCTION

6.5.1.1 The Onsite Review Committee shall function to advise the ~~Plant~~ Station Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Onsite Review Committee shall be composed of the:

Chairman:	~~Plant Manager~~ Station Manager
Member:	~~Plant Superintendent~~ Assistant Station Manager, Technic
Member:	~~Supervisor of Plant Operations~~ " " " Operatio
Member:	~~Technical Supervisor~~ " " "
Member:	~~Supervisor of Plant Maintenance~~ Health Physics Manager , Maintena
Member:	Plant Instrument and Control ~~Engineer~~ Supervisor
Member:	~~Plant Nuclear Engineer~~ Station Services Manager
Member:	~~Chemical and Radiation Protection Engineer~~
Member:	SDG&E Representative (Station Engineering Representati

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSRC Chairman (or designated Alternate) to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The OSRC shall meet at least once per calendar month and as convened by the OSRC Chairman or his designated alternate.

QUORUM

6.5.1.5 The minimum quorum of the OSRC necessary for the performance of the 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 Onsite Review Committee shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) all programs required by Specification 6.8 and changes thereto, 3) any other proposed procedures or changes thereto as determined by the ~~Plant~~ Station 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" Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence ~~to the Manager of Nuclear Operations and~~ to the Nuclear ~~Audit and Review Committee (NARC).~~ Control Board (NCB)

f. Review of events requiring 24-hour written notification to the Commission.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the ~~Plant~~ Station Manager or the ~~NARC.~~ NCB.

i. Review of the Security Plan and implementing procedures and shall submit recommended changes to the ~~NARC.~~ NCB.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the ~~NARC.~~ NCB.

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Note 19

~~k. Review of every unplanned onsite release of radioactive material to the environs including the preparation and forwarding of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence to the Manager of Nuclear Operations and to the Nuclear Audit and Review Committee (NARC).~~

Note 20 k. ~~l.~~ Review of changes to the PROCESS CONTROL PROGRAM, OFFSITE DOSE CALCULATION MANUAL & ~~and radwaste treatment systems.~~

l. Review of major changes to radwaste treatment systems.

AUTHORITY

6.5.1.7 The Onsite Review Committee (OSRC) shall:

Note 7

a. Recommend in writing to the Station ~~Plant~~ Manager approval or disapproval of items considered under 6.5.1.6(b ~~a~~) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(b ~~a~~) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to ~~the Manager of Nuclear Operation and~~ the Nuclear ~~Audit and Review Committee~~ Control Board of disagreement between the OSRC and the Station ~~Plant~~ Manager; however, the Station ~~Plant~~ Manager shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Onsite Review Committee shall maintain written minutes of each OSRC meeting that, at a minimum, document the results of all OSRC activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to ~~the Manager of Nuclear Operations and~~ the Nuclear ~~Audit and Review Committee.~~ Control Board.

Note 5

6.5.2 NUCLEAR SAFETY GROUP (NSG) ~~AUDIT AND REVIEW COMMITTEE (NARC)~~

FUNCTION

6.5.2.1 The Nuclear Safety Group (NSG) ~~Audit and Review Committee~~ shall function to provide independent review and audit of designated activities in the areas of:

a. nuclear power plant operations

b. nuclear engineering

c. chemistry and radiochemistry

d. metallurgy

e. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

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COMPOSITION

6.5.2.2 The ~~NARC~~ NSG shall be composed of ~~the:~~ a minimum of a supervisor and three staff specialists.

~~Member: Manager of Engineering Design~~
~~Member: Manager of Environmental Affairs~~
~~Member: Manager of Nuclear Engineering Safety & Licensing~~
~~Member: Manager, Quality Assurance~~
~~Member: Manager of Nuclear Operations~~
~~Member Manager, Nuclear Engineering and Safety~~
~~Member Manager, Biological Systems Research and Development~~
~~Member SDG&E Representative~~

~~Chairmanship shall be designated by the Nuclear Control Board.~~

~~ALTERNATES~~

~~6.5.2.3 All alternate members shall be appointed in writing by the NARC Director to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NARC activities at any one time.~~

CONSULTANTS

6.5.2.~~4~~3 Consultants shall be utilized as determined by the ~~NARC Director~~ NSG Supervisor to provide expert advice to the ~~NARC.~~ NSG.

~~MEETING FREQUENCY~~

~~6.5.2.5 The NARC 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 necessary for the performance of the NARC review and audit functions of these Technical Specifications shall consist of the Director or his designated alternate and (at least 4 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~~4 The ~~NARC~~ NSG shall review:

Note 7

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.

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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.

Note 9

i. Reports and meetings minutes of the Onsite Review Committee.

j. Results of audits conducted to meet the requirements of section 6.5.2.5 below.

AUDITS

6.5.2.~~8~~5 Audits of unit activities shall be performed under the cognizance of the ~~NARC.~~ Quality Assurance Organization. These audits shall encompass:

Note 9

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 NARC or~~ the Nuclear Control Board.

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.

k. The radiological environmental monitoring program and the results thereof at least once per 12 months.

l. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

m. The PROCESS CONTROL PROGRAM and implementing procedures for solidication of radioactive wastes at least once per 24 months.

n. The performance of activities required by the Quality Assurance Program to meet the criteria of Regulatory Guide 4.15, December 1977 at least once per 12 months.

AUTHORITY

6.5.2.~~9~~ 6 The ~~NARC~~ NSG shall report to and advise the ~~Nuclear Control Board~~ Manager, Nuclear Engineering and Safety on those areas of responsibility specified in Sections 6.5.2.~~7~~ 4 and 6.5.2.~~8~~ 5.

RECORDS

Note 10

6.5.2.~~10~~ 7 Records of ~~NARC~~ NSG activities shall be prepared~~, approved and~~ and maintained. A summary shall be distributed ~~as indicated below:~~ each calendar month to the Station Manager and NCB members.

~~a. Minutes of each NARC meeting shall be prepared, approved and forwarded to the Nuclear Control Board within 14 days following each meeting.~~

~~b. Reports of reviews encompassed by Section 6.5.2.7 above, shall be prepared, approved and forwarded to the Nuclear Control Board within 14 days following completion of the review.~~

~~c. Audit reports encompassed by Section 6.5.2.8 above, shall be forwarded to the Nuclear Control Board and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.~~

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NOTE 11

6.5.3 NUCLEAR CONTROL BOARD

FUNCTION

6.5.3.1 The Nuclear Control Board (NCB) shall function to provide company direction in the resolution of significant safety issues.

COMPOSITION

6.5.3.2 The NCB shall be composed of the:

Chairman: Vice President Nuclear Engineering and Operations

Member: Vice President Engineering and Construction

Member: Vice President Advanced Engineering

Member: Manager of Nuclear Engineering, Licensing, and Safety

Member: Manager of Nuclear Operations

Member: Manager, Nuclear Engineering and Safety

Member: Manager, Quality Assurance

Member: San Diego Gas and Electric Representative

ALTERNATES

6.5.3.3 All alternate members shall be appointed in writing by the NCB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in NCB activities at any one time.

MEETING FREQUENCY

6.5.3.4 The NCB shall meet at least once per six months.

QUORUM

6.5.3.5 A quorum of the NCB shall consist of the Chairman or his designated alternate and three members including alternates.

RESPONSIBILITIES

6.5.3.6 With respect to these Technical Specifications, the NCB shall oversee the activities of the NSG and OSRC and maintain management control for nuclear safety issues.

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~~6.6 REPORTABLE OCCURRENCE ACTION~~

~~6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:~~

Note 12

~~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 and submitted to the NARC and the Manager of Nuclear Operations.~~

6.~~7~~6 SAFETY LIMIT VIOLATION

6.~~7~~6.1 The following actions shall be taken in the event a Safety Limit is violated:

Note 13

a. The ~~unit shall be placed in at least HOT STANDBY within one hour.~~ action required by Specification 2.0 shall be taken,

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 and the ~~NARC~~ NCB Chairman shall be notified within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the OSRC. 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~~ and the ~~Manager of Nuclear Operations~~ NCB within 14 days of the violation.

6.~~8~~7 PROCEDURES AND PROGRAMS

6.~~8~~7.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

Note 14

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision ~~2~~ 1, February ~~1978~~ 1975.

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.

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g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Assurance Program for effluent and environmental monitoring, using the guidance in Regulatory Guide 4.15, December 1977.

see attached

~~6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the OSRC and approved by the Plant Manager 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:~~

Note 15

~~a. The intent of the original procedure is not altered.~~

see attached

~~b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.~~

~~c. The change is documented, reviewed by the OSRC and approved by the Plant Manager within 14 days of implementation.~~

6.~~8~~7.4 The following programs shall be established, implemented, and maintained:

a. Primary Coolant Sources Outside Containment

A program to reduce leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include ~~(~~the ~~recirculation~~ containment spray, safety injection, ~~chemical and volume control, gas stripper, and hydrogen recombiners)~~ and post-accident sampling. The program shall include the following:

Note: see Q&R 312.45

(i) Preventive maintenance and periodic visual inspection requirements, and

(ii) Integrated leak test requirements for each system at refueling cycle intervals or less.

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

(i) Training of personnel,

(ii) Procedures for monitoring, and

(iii) Provisions for maintenance of sampling and analysis equipment.

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6.~~8~~7.2 The initial issue of each procedure described in Specification 6.8.1 above shall be reviewed by two members of the OSRC (or alternates) to ensure that no change to the Technical Specification is required and no unreviewed safety question exists; and shall be approved by the cognizant supervisor prior to implementation.

a. Procedures which may affect the operational status of plant systems or equipment shall be reviewed by a Senior Reactor Operator.

b. Procedure revisions shall be reviewed at periodic intervals as set forth in administrative procedures.

c. For procedures which involve a deviation from the Technical Specifications or an unreviewed safety question, a safety evaluation shall be performed and approved by the OSRC and NSG. NRC approval shall be obtained prior to implementation.

6.~~8~~7.3 Revisions to procedures of 6.8.1 above which do not involve a deviation from the Technical Specification or an unreviewed safety question may be made provided:

a. Revisions which do not change the intent of the approved procedures, shall as a minimum be approved by two members of the plant staff knowledgeable in the areas affected by the procedure. At least one of these shall be a member of the OSRC (or an alternate).

b. Revisions which change the intent of the approved procedure, shall as a minimum be approved by two members of the OSRC (or alternate) knowledgeable in the areas affected by the procedure.

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c. Revisions to procedures which may affect the operational status of plant systems or equipment shall be approved by two members of the OSRC (or alternates) knowledgeable in the areas affected by the procedure and a Senior Reactor Operator if neither of the OSRC members is so qualified.

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c. Secondary Water Chemistry

A program for monitoring of secondary water chemistry to inhibit steam generator tube degradation. This program shall include:

(i) Identification of a sampling schedule for the critical variables and control points for these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, including monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Post-Accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the training of personnel, the procedures for sampling and analysis and the provisions for maintenance of sampling and analysis equipment

6.~~9~~8 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.~~9~~8.1 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~~8.1.1 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.

[redacted]

6.8.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.8.1.3 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*

6.8.1.4 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.

6.8.1.5 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,** e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total whole body dose received from external sources should be assigned to specific major work functions.

~~b. The results of the core barrel movement monitoring activities performed during the report period. (CE units only).~~

~~c. (Any other unit unique reports required on an annual basis.)~~

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

** This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.8.1.6 Routine radiological environmental operating reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

6.8.1.7 The annual radiological environmental operating reports shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2. If harmful effects or evidence of irreversible damage are detected by the monitoring, the report shall provide an analysis of the problem and a planned course of action to alleviate the problem.

The annual radiological environmental operating reports shall include summarized and tabulated results in the format of Regulatory Guide 4.8, December 1975 of all radiological environmental samples taken during the report period. In the event that some results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; a map of all sampling locations keyed to a table giving distances and directions from one reactor; and the results of licensee participation in the Interlaboratory Comparison Program, required by Specification 3.12.3.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.8.1.8 Routine radioactive effluent release reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

*A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

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6.~~9~~8.1.9 The radioactive effluent release reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing of wind speed, wind direction, and atmospheric stability, and precipitation (if measured) on magnetic tape, or in the form of stability. This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary (Figures 5.1-3 and 5.1-4) during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time and location) shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents (as determined by sampling frequency and measurement) shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the OFFSITE DOSE CALCULATION MANUAL (ODCM).

The radioactive effluent release report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other nearby uranium fuel cycle sources (including doses from primary effluent pathways and direct radiation) for the previous 12 consecutive months to show conformance with 40 CFR 190, Environmental Radiation Protection Standards for Nuclear Power Operation. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Rev. 1.

The radioactive effluents release shall include the following information for each type of solid waste shipped offsite during the report period:

a. Container volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms),

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent (e.g., cement, urea formaldehyde).

The radioactive effluent release reports shall include unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents on a quarterly basis.

The radioactive effluent release reports shall include any changes to the PROCESS CONTROL PROGRAM (PCP) made during the reporting period.

MONTHLY OPERATING REPORT

6.~~9~~8.1.10 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the safety valves, shall be submitted on a monthly basis to the Director, Office of Management and Program Analysis, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Office of Inspection and Enforcement, no later than the 15th of each month following the calendar month covered by the report.

Any changes to the OFFSITE DOSE CALCULATION MANUAL shall be submitted with the Monthly Operating Report within 90 days in which the change(s) was made effective. In addition, a report of any major changes to the radioactive waste treatment systems shall be submitted with the Monthly Operating Report for the period in which the evaluation was reviewed and accepted by the ~~(Unit Review Group).~~ Onsite Review Committee

REPORTABLE OCCURRENCES

6.~~9~~8.1.7 The REPORTABLE OCCURRENCES of Specifications 6.~~9~~8.1.8 and 6.~~9~~8.1.9 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~~8.1.8 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.

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.

j. Offsite releases of radioactive materials in liquid and gaseous effluents which exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

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k. Exceeding the limits in Specification 3.11.1.4 or 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written follow-up report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY DAY WRITTEN REPORTS

6.~~9~~8.1.9 The types of events listed below shall be the subject of written reports to the Director of the Regional Office within thirty days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety feature instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety feature systems.

d. Abnormal degradation of systems other than those specified in 6.9.1.8.c above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of 1) more than 1 curie of radioactive material in liquid effluents, 2) more than 150 curies of noble gas in gaseous effluents, or 3) more than 0.05 curies of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved.

2. Cause(s) for the unplanned release.

3. Actions taken to prevent recurrence.

4. Consequences of the unplanned release.

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f. Measured levels of radioactivity in an environmental sampling medium determined to exceed the reporting level values of Table 3.12-2 when averaged over any calendar quarter sampling period.

SPECIAL REPORTS

6.~~9~~8.2 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~~9 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~~9.1 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, reactor ~~tests~~ and experiments.

f. Records of radioactive shipments.

Note 16

~~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~~9.2 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.

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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.

Note 18 ~~i. Records of Quality Assurance activities required by the QA Manual.~~

Note 17 ~~j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.~~

i. ~~k.~~ Records of ~~meetings of the~~ OSRC meetings and ~~the NARC.~~ NSG reviews

Note 15 ~~l. Records of the service lives of all snubbers listed in Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.~~

~~m. Records of secondary water sampling and water quality.~~

j. ~~n.~~ Records of analyses required by the radiological environmental monitoring program.

6.~~11~~ 10 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~~ 11 HIGH RADIATION AREA ~~(OPTIONAL)~~

6.~~12~~ 11.1 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

Note 18 6.9.3 Other records required by the Quality Assurance Manual shall be retained for a period of time in accordance with the guidance of ANSI N45.2.9-1974.

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Exposure Permit (REP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. 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~~11.2 The requirements of 6.~~12~~11.1, 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 Watch Engineer.

6.~~13~~12 PROCESS CONTROL PROGRAM (PCP)

6.~~13~~12.1 The PCP shall be approved by the Commission prior to implementation.

6.~~13~~12.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the semi-annual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to ~~totally~~ support the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

*Health Physics personnel or personnel escorted by Health Physics personnel shall be exempt from the RWP issuance requirement during the performance of their assigned radiation protection duties, provided they comply with approved radiation protection procedures for entry into high radiation areas.

c. Documentation of the fact that the change has been reviewed and found acceptable by the OSRC.

2. Shall become effective upon review and acceptance by the OSRC.

6.~~14~~13 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.~~14~~13.1 The ODCM shall be approved by the Commission prior to implementation.

6.~~14~~13.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Monthly Operating Report within 90 days of the date the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to ~~totally~~ support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the OSRC.

2. Shall become effective upon review and acceptance by the OSRC.

6.~~15~~14 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS (Liquid, Gaseous and solid)

6.~~15~~14.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Monthly Operating Report for the period in which the evaluation was reviewed by the OSRC. The discussion of each change shall contain:

   a. A summary of the evaluation that led to the detemrination that the change could be made in accordance with 10 CFR 50.59;

   b. Sufficient detailed information to ~~totally~~ support the reason for the change without benefit of additional or supplemental information;

c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

e. An evaluation of the change which shows the expected maximum exposures to individual in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

g. An estimate of the exposure to plant operating personnel as a result of the change; and

h. Documentation of the fact that the change was reviewed and found acceptable by the OSRC.

2. Shall become effective upon review and acceptance by the OSRC.