ST. LUCIE PLANT

UNIT 2

TECHNICAL SPECIFICATIONS

APPENDIX “A”

TO

LICENSE NO. NPF-16

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1.0 DEFINITIONS

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

ACTION

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

AXIAL SHAPE INDEX

1.2 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = AY_E + B$$

AZIMUTHAL POWER TILT - T_q

1.3 AZIMUTHAL POWER TILT shall be the maximum difference between the power generated in any core quadrant (upper or lower) and the average power of all quadrants in that half (upper or lower) of the core divided by the average power of all quadrants in that half (upper or lower) of the core.

$$\text{Azimuthal Power Tilt} = \text{MAX}\left[\frac{\text{Power in any core quadrant (upper or lower)}}{\text{Average power of all quadrants (upper or lower)}}\right] - 1$$

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.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CONTAINMENT VESSEL INTEGRITY

1.7 CONTAINMENT VESSEL INTEGRITY shall exist when:

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

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

2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except for valves that are open on an intermittent basis under administrative control.

b. All containment vessel equipment hatches are closed and sealed,

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

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

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

CONTROLLED LEAKAGE

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

CORE ALTERATION

1.9 CORE ALTERATION shall be the movement or manipulation of any fuel, sources, reactivity control components, or other components affecting reactivity within the reactor vessel with the vessel head removed and fuel in the vessel. Exceptions to the above include shared (4 fingered) control element assemblies (CEAs) withdrawn into the upper guide structure (UGS) or evolutions performed with the UGS in place such as CEA latching/unlatching or verification of latching/unlatching which do not constitute a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

1.9a The COLR is the unit-specific document that provides cycle specific parameter limits for the current operating reload cycle. These cycle-specific parameter limits shall be determined for each reload cycle in accordance with Specification 6.9.1.11. Plant operation within these limits is addressed in individual Specifications.

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 ICRP-30, Supplement to Part 1, pages 192-212, Tables entitled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity (Sv/Bq)."

$\bar{E}$ - 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 FEATURES RESPONSE TIME

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

FREQUENCY NOTATION

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

GASEOUS RADWASTE TREATMENT SYSTEM

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

IDENTIFIED LEAKAGE

1.15 IDENTIFIED LEAKAGE shall be:

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

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

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

LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE - RCS

1.16 The LOW TEMPERATURE OVERPRESSURE PROTECTION RANGE is that operating condition when (1) the RCS cold leg temperature is less than or equal to that specified in Table 3.4-3, and (2) the Reactor Coolant System is not vented to containment by an opening of at least 3.58 square inches.

MEMBER(S) OF THE PUBLIC

1.17 MEMBER OF THE PUBLIC means an individual in a controlled or unrestricted area. However, an individual is not a member of the public during any period in which the individual receives an occupational dose.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8.4 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Annual Radioactive Effluent Release Reports required by Specifications 6.9.1.7 and 6.9.1.8.

OPERABLE - OPERABILITY

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

OPERATIONAL MODE - MODE

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

PHYSICS TESTS

1.21 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

1.22 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)

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

PURGE - PURGING

1.24 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.

RATED THERMAL POWER

1.25 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2700 MWt.

REACTOR TRIP SYSTEM RESPONSE TIME

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

REPORTABLE EVENT

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

SHIELD BUILDING INTEGRITY

1.28 SHIELD BUILDING INTEGRITY shall exist when:

a. Each door is closed except when the access opening is being used for normal transit entry and exit;

b. The shield building ventilation system is in compliance with Specification 3.6.6.1, and

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

SHUTDOWN MARGIN

1.29 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length 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.

SITE BOUNDARY

1.30 Site Boundary means that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOURCE CHECK

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

STAGGERED TEST BASIS

1.32 A STAGGERED TEST BASIS shall consist of:

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

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

THERMAL POWER

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

UNIDENTIFIED LEAKAGE

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

UNRESTRICTED AREA

1.35 Unrestricted area means an area, access to which is neither limited nor controlled by the licensee.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.36 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in the unrodded core, excluding tilt.

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.37 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding tilt.

VENTILATION EXHAUST TREATMENT SYSTEM

1.38 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 Features (ESF) atmospheric cleanup systems are not considered to be VENTILATION EXHAUST TREATMENT SYSTEM components.

TABLE 1.1

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
4/M*	At least 4 per month at intervals of no greater than 9 days and a minimum of 48 per year.
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.

*For Radioactive Effluent Sampling.

**For Radioactive Batch Releases only.

TABLE 1.2

OPERATIONAL MODES

OPERATIONAL MODE	REACTIVITY CONDITION, K_{eff}	% OF RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	$\geq$ 0.99	> 5%	$\geq$ 325°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 325°F
3. HOT STANDBY	< 0.99	0	$\geq$ 325°F
4. HOT SHUTDOWN	< 0.99	0	325°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.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

2.1.1 REACTOR CORE

DNBR

2.1.1.1 The combination of THERMAL POWER, pressurizer pressure, and maximum cold leg coolant temperature shall not exceed the limits shown on Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the combination of THERMAL POWER, pressurizer pressure and maximum cold leg coolant temperature has exceeded the limits shown on Figure 2.1-1, be in HOT STANDBY within 1 hour, and comply with the requirements of Specification 6.7.1

REACTOR COOLANT SYSTEM PRESSURE

2.1.2 The Reactor Coolant System pressure shall not exceed 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, and comply with the requirements of Specification 6.7.1.

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, and comply with the requirements of Specification 6.7.1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

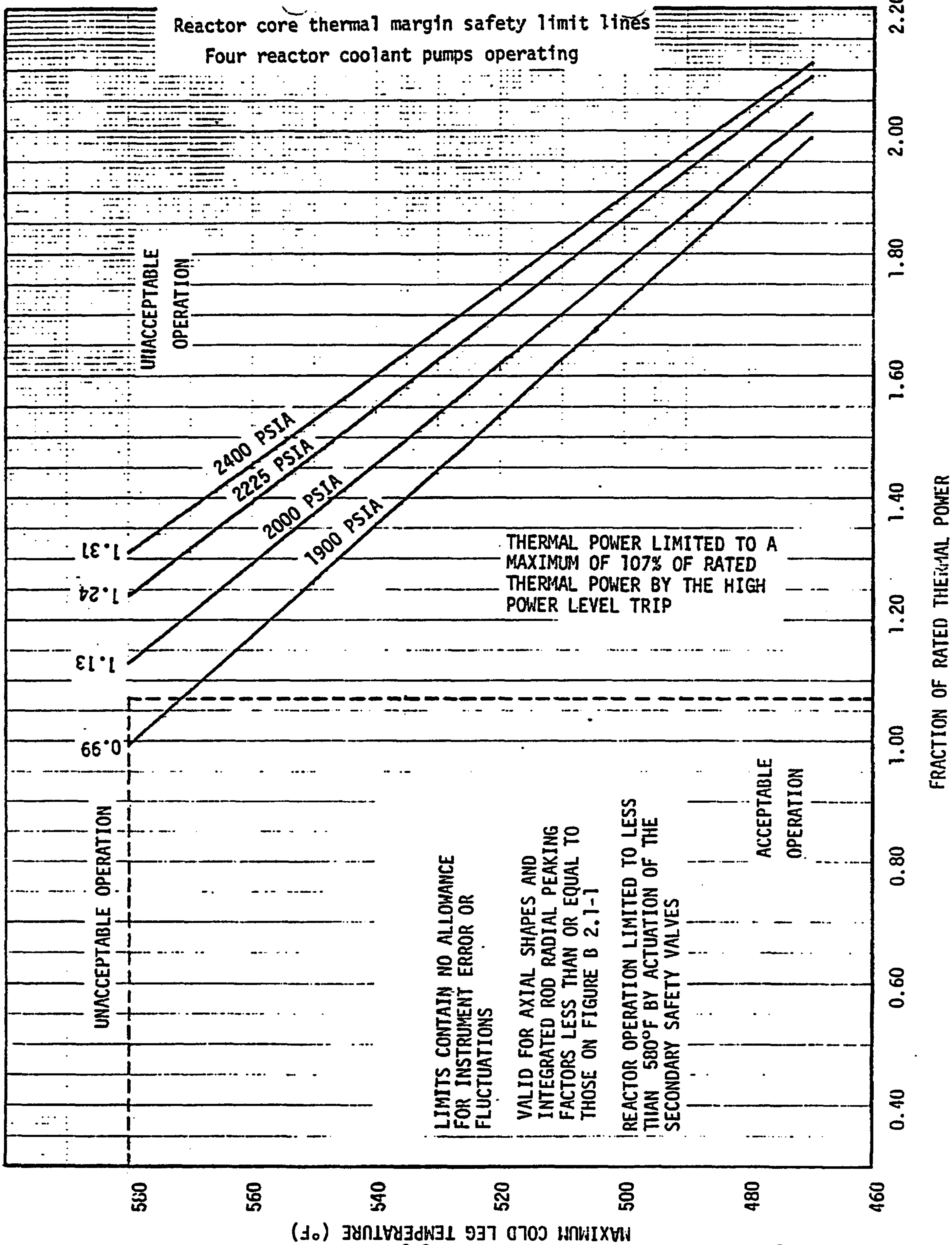
2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

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.

Figure 2.1-1



Reactor core thermal margin safety limit lines

Four reactor coolant pumps operating

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. Variable Power Level - High (1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Pressurizer Pressure - High	≤ 2370 psia	≤ 2374 psia
4. Thermal Margin/Low Pressure(1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4. Minimum value of 1900 psia.
5. Containment Pressure - High	≤ 3.0 psig	≤ 3.1 psig
6. Steam Generator Pressure - Low	≥ 626.0 psia (2)	≥ 621.0 psia (2)
7. Steam Generator Pressure(1) Difference - High (Logic in TM/LP Trip Unit)	≤ 120.0 psid	≤ 132.0 psid
8. Steam Generator Level - Low	≥ 20.5% (3)	≥ 19.5% (3)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density – High[(5)] Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.
10. Loss of Component Cooling Water to Reactor Coolant Pumps – Low	≥ 636 gpm**	≥ 636 gpm
11. Reactor Protection System Logic	Not Applicable	Not Applicable
12. Reactor Trip Breakers	Not Applicable	Not Applicable
13. Rate of Change of Power – High[(4)]	≤ 2.49 decades per minute	≤ 2.49 decades per minute
14. Reactor Coolant Flow – Low[(1)]	≥ 95.4% of design Reactor Coolant flow with four pumps operating*	≥ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure – Low[(5)]	≥ 800 psig	≥ 800 psig

* Design reactor coolant flow with four pumps operating is 355,000 gpm.

** 10-minute time delay after relay actuation.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

TABLE NOTATION

(1) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER during testing pursuant to Special Test Exception 3.10.3; bypass shall be automatically removed when the Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(3) % of the narrow range steam generator level indication.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

(5) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

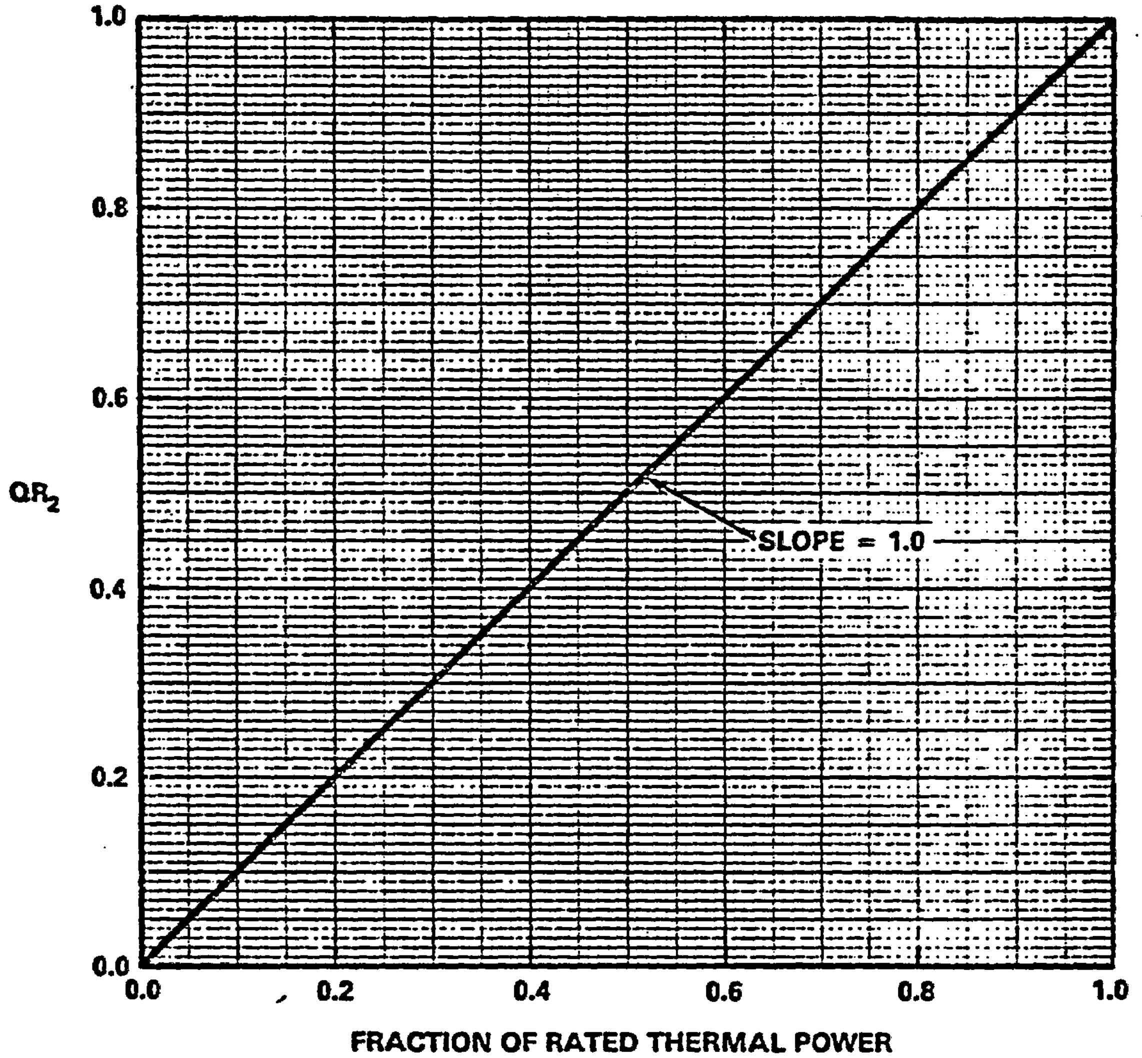


Figure 2.2-1
Local power density - High trip setpoint
Part 1 (Fraction of RATED THERMAL POWER versus QR_2)

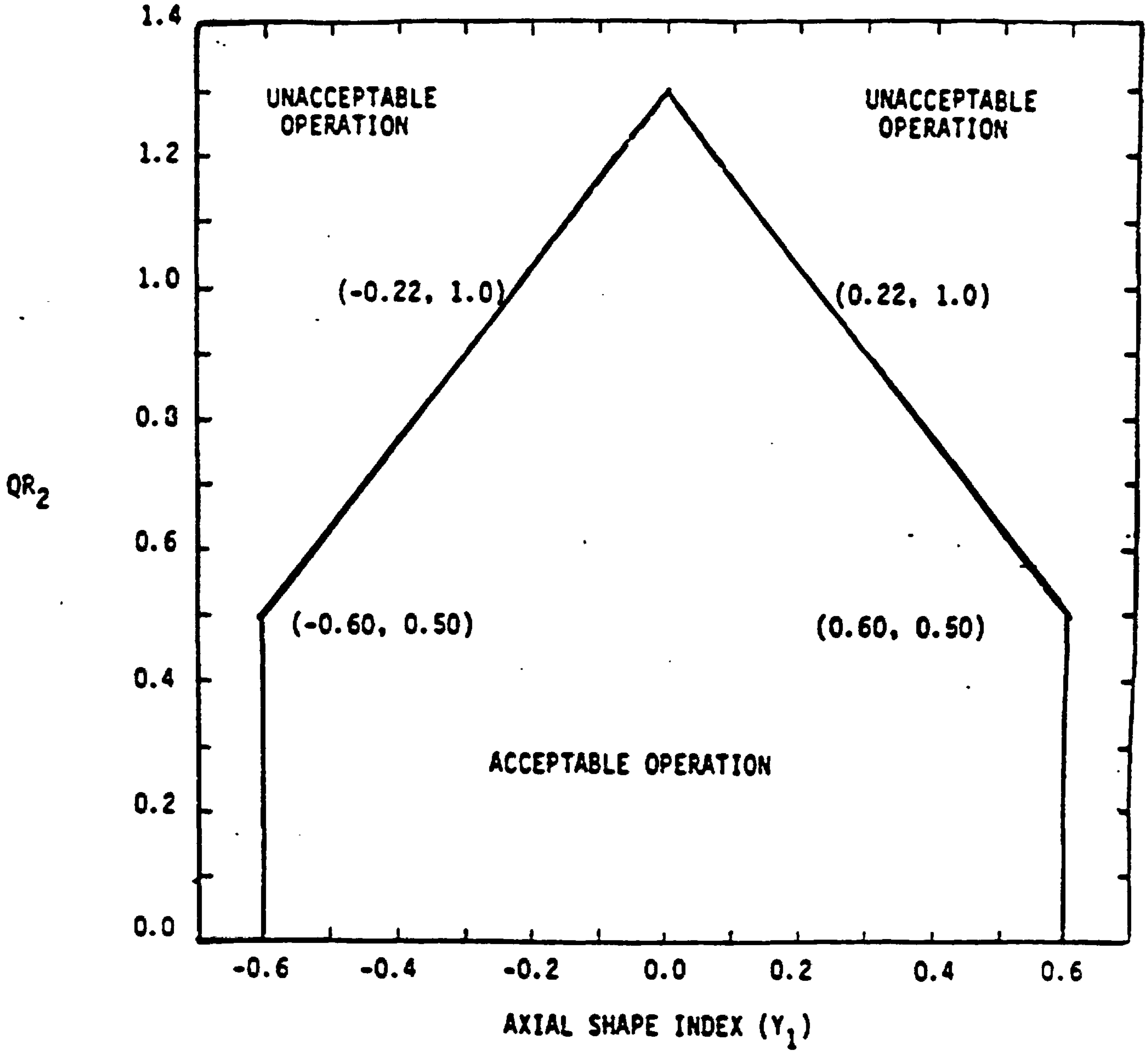


FIGURE 2.2-2

LOCAL POWER DENSITY-HIGH TRIP SETPOINT
PART 2 (QR_2 versus Y_1)

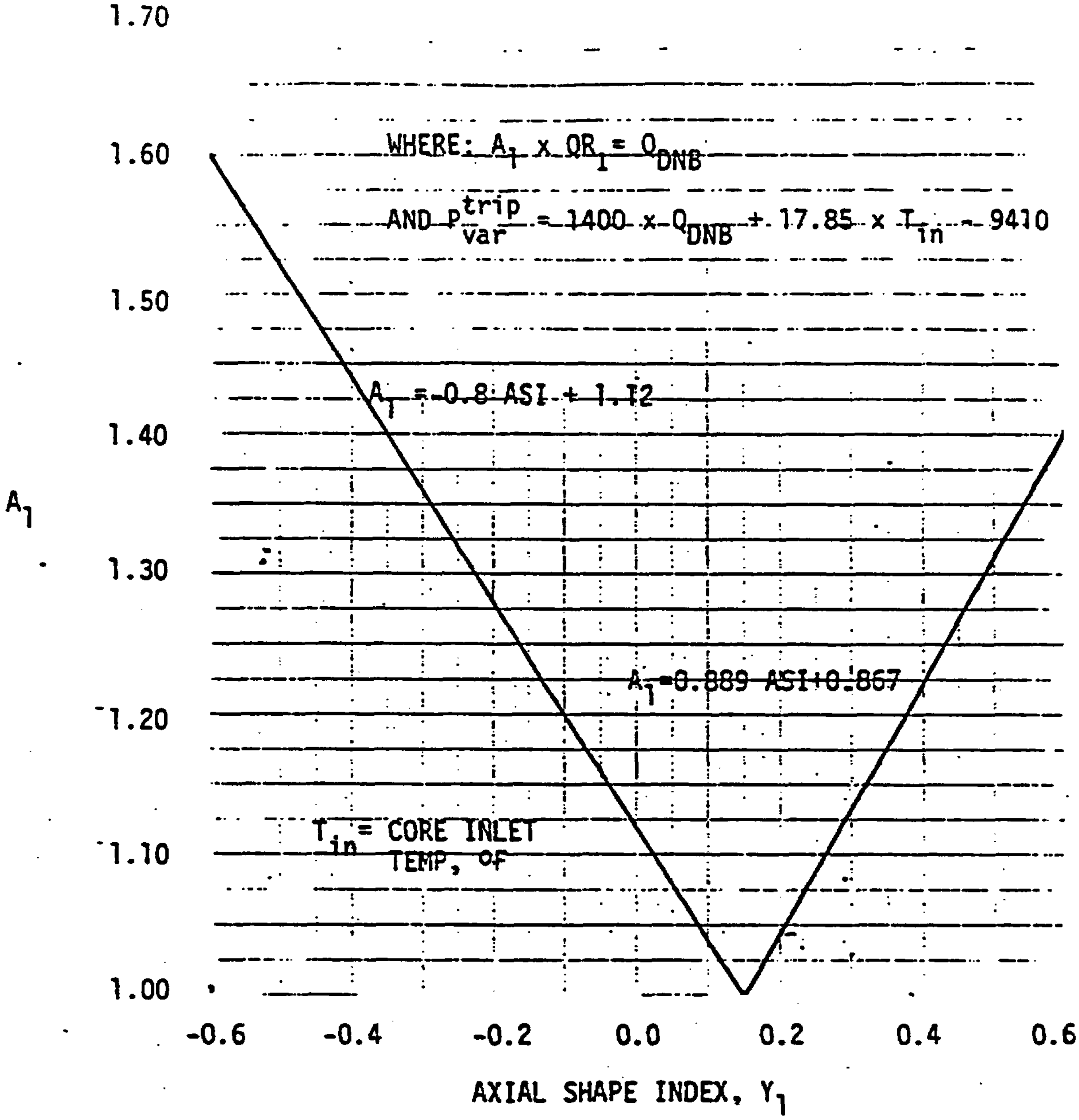


FIGURE 2.2-3

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 1 (Y_1 Versus A_1)

WHERE: $A_1 \times QR_1 = Q_{DNB}$

AND $P_{var}^{trip} = 1400 \times Q_{DNB} + 17.85 \times T_{in} - 9410$

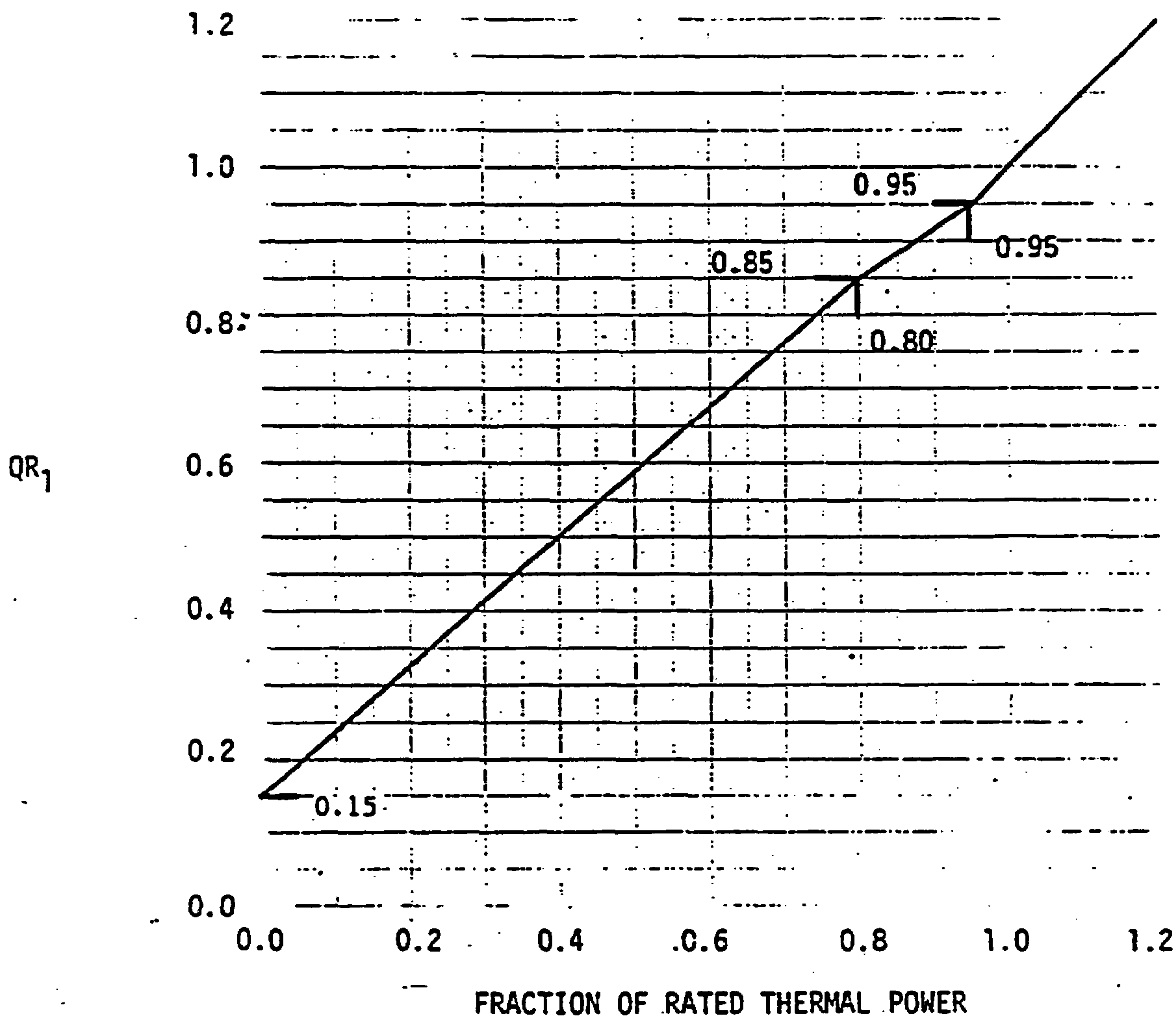


FIGURE 2.2-4

THERMAL MARGIN/LOW PRESSURE TRIP SETPOINT
PART 2 (FRACTION OF RATED THERMAL POWER VERSUS QR_1)

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4 LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

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

3.0.2 Noncompliance with a specification shall exist when the requirements of the Limiting Condition for Operation and/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 1 hour, action shall be initiated to place the unit in a MODE in which the specification does not apply by placing it, as applicable, in:

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

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

This specification is not applicable in MODE 5 or 6.

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

APPLICABILITY

SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance Requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement. Failure to perform a Surveillance Requirement within the allowable surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. Surveillance Requirements do not have to be performed on inoperable equipment.

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

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

If the Surveillance is not performed within the delay period, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(s) must be taken.

When the Surveillance is performed within the delay period and the Surveillance is not met, the Limiting Condition for Operation must immediately be declared not met, and the applicable ACTION(s) must be taken.

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

4.0.5 Surveillance Requirements for inservice inspection 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 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. deleted

c. deleted

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

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

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

3/4.1.1 BORATION CONTROL

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

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

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

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, 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 within the COLR limits:

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 not fully inserted, and is immovable as a result of excessive friction or mechanical interference or is known to be 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 Keff greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2 with Keff less than 1.0, within 4 hours prior to achieving reactor criticality by verifying that the predicted critical CEA position is within the limits of Specification 3.1.3.6.

* See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

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

e. When in MODE 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 $\pm$ 1000 pcm at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 EFPDs after each fuel loading.

REACTIVITY CONTROL SYSTEMS

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

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limits specified in the COLR.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN outside the COLR limits, 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 within the COLR limits:

a. Within 1 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.

c. At least once per 24 hours, when the Reactor Coolant System is drained below the hot leg centerline, by consideration of the factors in 4.1.1.2b and by verifying at least two charging pumps are rendered inoperable by racking out their motor circuit breakers.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be ≥ 3000 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: ALL MODES.

ACTION:

With the flow rate of reactor coolant to the reactor pressure vessel < 3000 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be determined to be ≥ 3000 gpm within 1 hour prior to the start of and at least once per hour during a reduction in the Reactor Coolant System boron concentration by either:

a. Verifying at least one reactor coolant pump is in operation, or

b. Verifying that at least one low pressure safety injection pump is in operation and supplying ≥ 3000 gpm to the reactor pressure vessel.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be maintained within the limits specified in the COLR. The maximum positive limit shall be:

a. Less positive than +5 pcm/°F at ≤ 70% RATED THERMAL POWER, and

b. Less positive than +3 pcm/°F at > 70% 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

4.1.1.4.1 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.

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

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

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 800 ppm.

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

*With K_{eff} greater than or equal to 1.0.
#See Special Test Exceptions 3.10.2 and 3.10.5.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be greater than or equal to 515°F.

APPLICABILITY: MODES 1 and 2#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) less than 515°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.5 The Reactor Coolant System temperature (T_{avg}) shall be determined to be greater than or equal to 515°F:

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

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 525°F.

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

REACTIVITY CONTROL SYSTEMS

3/4.1.2 BORATION SYSTEMS

FLOW PATHS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

b. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F by verifying that the Boric Acid Makeup Tank solution temperature is greater than 55°F (when the flow path from the Boric Acid Makeup Tank is used).

---

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

FLOW PATHS – OPERATING

LIMITING CONDITION FOR OPERATION

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

a. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a boric acid makeup pump through a charging pump to the Reactor Coolant System.

b. One flow path from the boric acid makeup tank(s) with the tank meeting Specification 3.1.2.8 part a) or b), via a gravity feed valve through a charging pump to the Reactor Coolant System.

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

OR

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

a. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both boric acid makeup pumps through a charging pump to the Reactor Coolant System.

b. One flow path from each boric acid makeup tank with the combined tank contents meeting Specification 3.1.2.8 c), via both gravity feed valves through a charging pump to the Reactor Coolant System.

c. 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 its COLR limit 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.

FLOW PATHS - OPERATING

SURVEILLANCE REQUIREMENTS

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

a. At least once per 24 hours, when the Reactor Auxiliary Building air temperature is below 55°F, by verifying that the solution temperature of the Boric Acid Makeup Tanks is above 55°F.

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 an SIAS test signal.

d. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a and 3.1.2.2b delivers at least 40 gpm to the Reactor Coolant System.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or 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 power source.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes*.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least the above required pumps shall be demonstrated OPERABLE by verifying the charging pump develops a flow rate of greater than or equal to 40 gpm or the high pressure safety injection pump develops a total head of greater than or equal to 2854 ft. when tested pursuant to the Inservice Testing Program.

---

* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

CHARGING PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 At least two charging pumps shall be demonstrated OPERABLE by verifying that each pump develops a flow rate of greater than or equal to 40 gpm when tested pursuant to the Inservice Testing Program.

4.1.2.4.2 At least once per 18 months verify that each charging pump starts automatically on an SIAS test signal.

BORIC ACID MAKEUP PUMPS – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.2.5 The above required boric acid makeup pump shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump develops a discharge pressure of greater than or equal to 90 psig when tested pursuant to the Inservice Testing Program.

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* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

BORIC ACID MAKEUP PUMPS – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid makeup pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2 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 3.1.2.2 is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2 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 its COLR limit at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump(s) develop a discharge pressure of greater than or equal to 90 psig when tested pursuant to the Inservice Testing Program.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES – SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. One boric acid makeup tank with a minimum borated water volume of 3550 gallons of 2.5 to 3.5 weight percent boric acid (4371 to 6119 ppm boron).

b. The refueling water tank with:

1. A minimum contained borated water volume of 125,000 gallons,

2. A minimum boron concentration of 1720 ppm, and

3. A solution temperature between 40°F and 120°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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 outside the range of 40°F and 120°F.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F, by verifying that the boric acid makeup tank solution temperature is greater than 55°F when that boric acid makeup tank is required to be OPERABLE.

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* Plant temperature changes are allowed provided the temperature change is accounted for in the calculated SHUTDOWN MARGIN.

BORATED WATER SOURCES – OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.8 At least two of the following four borated water sources shall be OPERABLE:

a. Boric Acid Makeup Tank 2A in accordance with Figure 3.1-1.

b. Boric Acid Makeup Tank 2B in accordance with Figure 3.1-1.

c. Boric Acid Makeup Tanks 2A and 2B with a minimum combined contained borated water volume in accordance with Figure 3.1-1.

d. The refueling water tank with:

1. A minimum contained borated water volume of 417,100 gallons,

2. A boron concentration of between 1720 and 2100 ppm of boron, and

3. A solution temperature of between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank(s) inoperable, restore the tank(s) 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 its COLR limit at 200°F; restore the above required boric acid makeup tank(s) 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 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

4.1.2.8 At least two required borated water sources shall be demonstrated OPERABLE:

a. At least once per 7 days by:

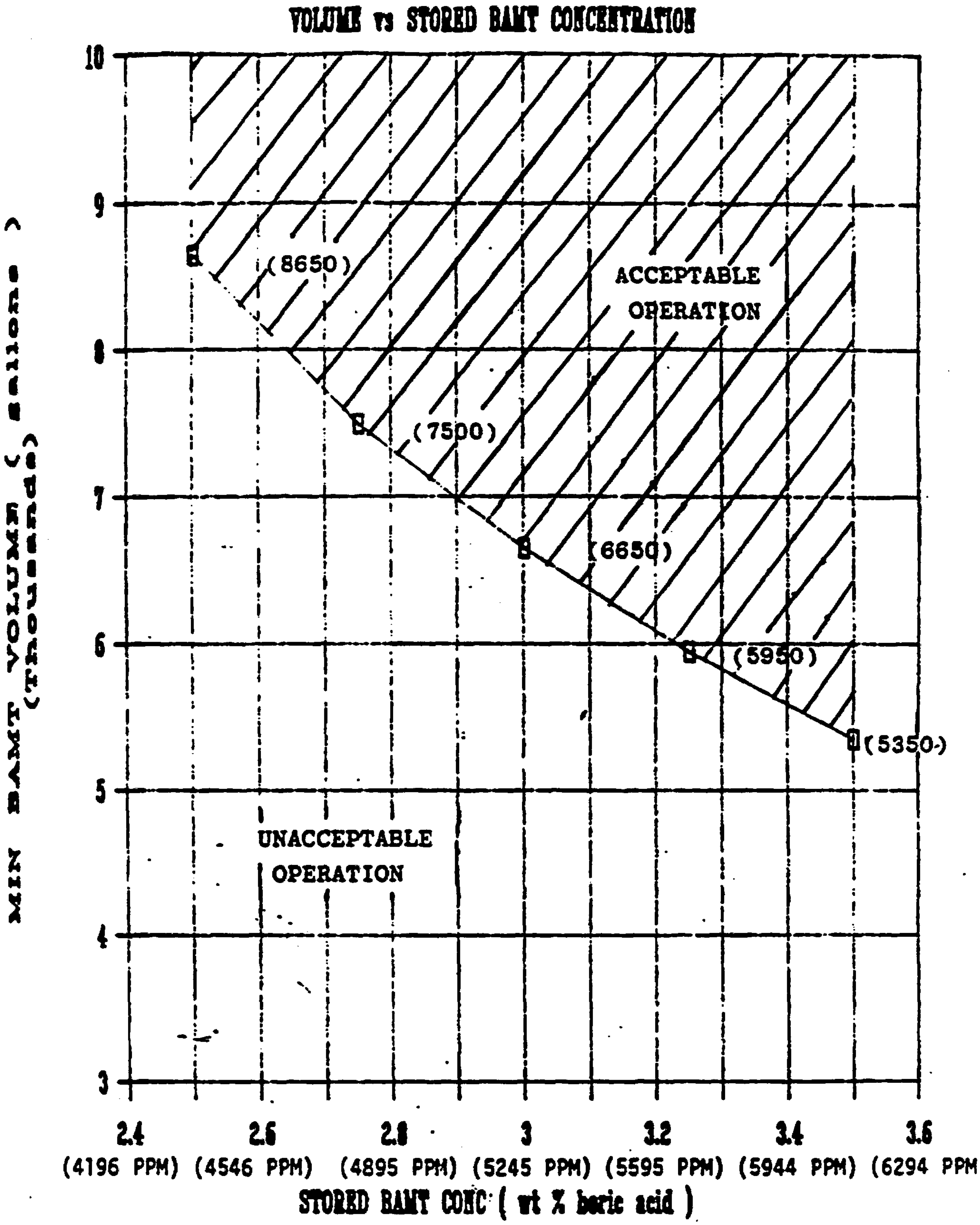
1. Verifying the boron concentration in the water and

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

b. At least once per 24 hours by verifying the RWT temperature when the outside air temperature is outside the range of 55°F and 100°F.

c. At least once per 24 hours when the Reactor Auxiliary Building air temperature is less than 55°F, by verifying that the boric acid makeup tank solution is greater than 55°F.

FIGURE 3.1-1 ST. LUCIE 2 MIN BAMT



Page 3/4 1-17 (Amendment No. 8) has been deleted from the Technical Specifications.
The next page is 3/4 1-18.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Block Circuit and all full-length (shutdown and regulating) CEAs which are inserted in the core, shall be OPERABLE with each CEA of a given group positioned within 7.0 inches (indicated position) of all other CEAs in its group.

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

b. With the CEA Block Circuit inoperable, within 6 hours either:

   1. With one CEA position indicator per group inoperable take action per Specification 3.1.3.2, or

   2. With the group overlap and/or sequencing interlocks inoperable maintain CEA groups 1, 2, 3 and 4 fully withdrawn and the CEAs in group 5 to less than 15% insertion and place and maintain CEA drive system in either the "Manual" or "Off" position, or

   3. Be in at least HOT STANDBY.

c. With more than one full-length CEA inoperable or misaligned from any other CEA in its group by more than 15 inches (indicated position), be in at least HOT STANDBY within 6 hours.

d. With one full-length CEA misaligned from any other CEA in its group by more than 15 inches, operation in MODES 1 and 2 may continue, provided that the misaligned CEA is positioned within 15 inches of the other CEAs in its group in accordance with the time constraints shown in COLR Figure 3.1-1a.

*See Special Test Exceptions 3.10.2, 3.10.4 and 3.10.5.

ACTION: (Continued)

e. With one full-length CEA misaligned from any other CEA in its group by more than 15 inches beyond the time constraints shown in COLR Figure 3.1-1a, reduce power to ≤ 70% of RATED THERMAL POWER prior to completing ACTION e.1 or e.2.

   1. Restore the CEA to OPERABLE status within its specified alignment requirements, or

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

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.0 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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

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

f. With one or more full-length CEA(s) misaligned from any other CEAs in its group by more than 7.0 inches but less than or equal to 15 inches, operation in MODES 1 and 2 may continue, provided that within 1 hour the misaligned CEA(s) is either:

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

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

      a) Within 1 hour the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.0 inches of the inoperable CEA while maintaining the allowable CEA sequence and insertion limits shown on COLR Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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

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

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

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* If the pre-misalignment ASI was more negative than -0.15, reduce power to ≤ 70% of RATED THERMAL POWER or 70% of the THERMAL POWER level prior to the misalignment, whichever is less, prior to completing ACTION e.2.a) and e.2.b).

ACTION: (Continued)

h. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in full-length CEA group 5, operation in MODES 1 and 2 may continue.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The Position of each full-length CEA shall be determined to be within 7.0 inches (indicated position) of all other CEAs in its group at least once per 12 hours except during time intervals when the Deviation Circuit and/or CEA Block Circuit 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 in the core shall be determined to be OPERABLE by movement of at least 7.0 inches in any one direction at least once per 92 days.

4.1.3.1.3 The CEA Block Circuit shall be demonstrated OPERABLE at least once per 92 days by a functional test which verifies that the circuit prevents any CEA from being misaligned from all other CEAs in its group by more than 7.0 inches (indicated position).

4.1.3.1.4 The CEA Block Circuit shall be demonstrated OPERABLE by a functional test which verifies that the circuit maintains the CEA group overlap and sequencing requirements of Specification 3.1.3.6 and that the circuit prevents the regulating CEAs from being inserted beyond the Power Dependent Insertion Limit of COLR Figure 3.1-2:

*a. Prior to each entry into MODE 2 from MODE 3, except that such verification need not be performed more often than once per 92 days, and

b. At least once per 6 months.

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* The licensee shall be excepted from compliance during the initial startup test program for an entry into MODE 2 from MODE 3 made in association with a measurement of power defect.

POSITION INDICATOR CHANNELS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and regulating CEA reed switch position indicator channels and CEA pulse counting position indicator channels shall be OPERABLE and capable of determining the absolute CEA positions within ± 2.50 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one reed switch position indicator channel per group or one (except as permitted by ACTION item c. below) pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel partially inserted, within 6 hours either:

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

   2. Be in HOT STANDBY, or

   3. Reduce THERMAL POWER to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant pump combination; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Operation at or below this reduced THERMAL POWER level may continue provided that within the next 4 hours either:

      a) The CEA group(s) with the inoperable position indicator is fully withdrawn while maintaining the withdrawal sequence required by Specification 3.1.3.6 and when this CEA group reaches its fully withdrawn position, the "Full Out" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully withdrawn. Subsequent to fully withdrawing this CEA group(s), the THERMAL POWER level may be returned to a level consistent with all other applicable specifications; or

      b) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

POSITION INDICATOR CHANNELS - OPERATING

ACTION: (Continued)

b. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the full-length CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

   1. The position of an affected full-length CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable), and

   2. The fully inserted full-length CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted, and

   3. Subsequent operation is within the limits of Specification 3.1.3.6.

c. With two or more pulse counting position indicators channels per group inoperable, operation in MODES 1 and 2 may continue for up to 72 hours provided no more than one reed switch position indicator per group is inoperable.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 5.0 inches at least once per 12 hours except during time intervals when the Deviation Circuit is inoperable, then compare the pulse counting position indicator and reed switch position indicator channels at least once per 4 hours.

REACTIVITY CONTROL SYSTEMS

POSITION INDICATOR CHANNELS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.3.3 At least one CEA position indicator channel shall be OPERABLE for each shutdown or regulating 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 position 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.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full-length (shutdown and regulating) CEA drop time, from a fully withdrawn position, shall be less than or equal to 3.1 seconds from when the electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90% insertion position with:

a. T_{avg} greater than or equal to 515°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:

   1. If in MODE 1 or 2, be in at least HOT STANDBY within 6 hours, or

   2. If in MODE 3, 4, or 5, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

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

SURVEILLANCE REQUIREMENTS

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

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

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

c. At least once per 18 months:

SHUTDOWN CEA INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All shutdown CEAs shall be withdrawn to greater than or equal to 129.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn to less than 129.0 inches, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 1 hour either:

a. Withdraw the CEA to greater than or equal to 129.0 inches, 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 greater than or equal to 129.0 inches:

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.

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*See Special Test Exception 3.10.2.

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

REGULATING CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating CEA groups shall be limited to the withdrawal sequence and to the insertion limits shown on COLR Figure 3.1-2 (regulating CEAs are considered to be fully withdrawn in accordance with COLR Figure 3.1-2 when withdrawn to greater than or equal to 129.0 inches), with CEA insertion between the Long Term Steady State Insertion Limits and the Power Dependent 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.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

a. With the regulating CEA groups inserted beyond the Power Dependent Insertion Limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, within 2 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 and insertion limits specified in the COLR .

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent 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 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, 3.10.4 and 3.10.5.

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

ACTION: (Continued)

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

   1. Restore the regulating groups to within the Long Term Steady State Insertion Limits within 2 hours, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating CEA group shall be determined to be within the Power Dependent 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 Power Dependent Insertion Limits shall be determined at least once per 24 hours.

3/4.2 POWER DISTRIBUTION LIMITS

3/4 2.1 LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits specified in the COLR.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of COLR Figure 3.2-2, 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 1 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 by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the linear heat rate by :

a. Verifying at least once per 12 hours that the full-length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limit shown on COLR Figure 3.2-2 .

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of COLR Figure 3.2-2, where 100% of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

M x N

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F^T_{xy} curve of COLR Figure 3.2-3 .

4.2.1.4 Incore Detector Monitoring System # - The incore detector monitoring system may be used for monitoring the linear heat rate by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on COLR Figure 3.2-1 .

If incore system becomes inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.3.

Pages 3/4 2-4 (Amendment 42), 3/4 2-5 (Amendment 8), and 3/4 2-6 (Amendment 17) have been deleted from the Technical Specifications. The next page is 3/4 2-7.

POWER DISTRIBUTION LIMITS

3/4.2.2 TOTAL PLANAR RADIAL PEAKING FACTORS - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F^T_{xy} shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} not within limits, within 6 hours either :

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of COLR Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+T_q)$ when F_{xy} is calculated with a non-full core power distribution analysis code and shall be calculated as $F^T_{xy} = F_{xy}$ when calculations are performed with a full core power distribution analysis code. F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70% of RATED THERMAL POWER after each fuel loading,

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

c. Within 4 hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing reactor coolant pump combination. This determination shall be limited to core planes between 15% and 85% of full core height and shall exclude regions influenced by grid effects.

4.2.2.4 T_q shall be determined each time a calculation of F^T_{xy} is made using a non full core power distribution analysis code. The value of T_q used in this case to determine F^T_{xy} shall be the measured value of T_q.

POWER DISTRIBUTION LIMITS

TOTAL INTEGRATED RADIAL PEAKING FACTORS - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T not within limits, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of COLR Figure 3.2-3 and withdraw the full-length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from COLR Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on COLR Figure 3.2-4 (truncate COLR Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by COLR Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of COLR Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ when F_r is calculated with a non-full core power distribution analysis code and shall be calculated as $F_r^T = F_r$ when calculations are performed with a full core power distribution analysis code. F_r^T shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70% of RATED THERMAL POWER after each fuel loading,

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

c. Within 4 hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing reactor coolant pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is made using a non-full core power distribution analysis code. The value of T_q used to determine F_r^T in this case shall be the measured value of T_q.

Page 3/4 2-12 (Amendment 42) has been deleted from the Technical Specifications. The next page is 3/4 2-13.

3/4.2.4 AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.03.

APPLICABILITY: MODE 1*.

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > .030 but ≤ .10, either correct the power tilt within 2 hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days.

b. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

*See Special Test Exception 3.10.2.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the limits shown on Table 3.2-2.

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-2 shall be verified to be within their limits by instrument readout at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement* at least once per 18 months.

---

* Not required to be performed until THERMAL POWER is ≥ 90% of RATED THERMAL POWER.

TABLE 3.2-2

DNB MARGIN

LIMITS

PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING
Cold Leg Temperature (Narrow Range)	535°F* $\leq$ T $\leq$ 549°F
Pressurizer Pressure	2225 psia** $\leq P_{PZR} \leq$ 2350 psia*
Reactor Coolant Flow Rate	$\geq$ 355,000 gpm
AXIAL SHAPE INDEX	COLR Figure 3.2-4

* Applicable only if power level $\geq$ 70% RATED THERMAL POWER.

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

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.

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. Neutron detectors are exempt from response time testing. 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.

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	4	2	4	1, 2	1
	4	2	4	3*, 4*, 5*	5
2. Variable Power Level - High	4	2(a)(d)	3	1, 2	2#
3. Pressurizer Pressure - High	4	2	3	1, 2	2#
4. Thermal Margin/Low Pressure	4	2(a)(d)	3	1, 2	2#
5. Containment Pressure - High	4	2	3	1, 2	2#
6. Steam Generator Pressure - Low	4/SG	2/SG(b)	3/SG	1, 2	2#
7. Steam Generator Pressure Difference - High	4	2(a)(d)	3	1, 2	2#
8. Steam Generator Level - Low	4/SG	2/SG	3/SG	1, 2	2#
9. Local Power Density - High	4	2(c)(d)	3	1	2#
10. Loss of Component Cooling Water to Reactor Coolant Pumps	4	2	3	1, 2	2#
11. Reactor Protection System Logic	4	2	3	1, 2	2#
				3*, 4*, 5*	5
12. Reactor Trip Breakers	4	2(f)	4	1, 2	4
				3*, 4*, 5*	5
13. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating - Rate of Change of Power - High	4	2(e)(g)	3	1, 2	2#
b. Shutdown	4	0	2	3, 4, 5	3
14. Reactor Coolant Flow - Low	4/SG	2/SG(a)(d)	3/SG	1, 2	2#
15. Loss of Load (Turbine Hydraulic Fluid Pressure - Low)	4	2(c)	3	1	2#

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 below 0.5% of RATED THERMAL POWER in conjunction with (d) below; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is greater than or equal to 0.5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 705 psig; bypass shall be automatically removed at or above 705 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when Power Range Neutron Flux power is greater than or equal to 15% of RATED THERMAL POWER.

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

(e) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when Wide Range Logarithmic Neutron Flux power is $\geq 10^{-4}$% and Power Range Neutron Flux power $\leq$ 15% of RATED THERMAL POWER.

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

(g) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

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 STATEMENTS

ACTION 2 - a. With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6m. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue provided the following conditions are satisfied:

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

2. All functional units affected by the bypassed/tripped channel shall also be placed in the bypassed/tripped condition.

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. Safety Channel - Nuclear Instrumentation	
Wide Range	Rate of Change of Power-High (RPS)
Linear Range	Variable Power Level-High (RPS) Local Power Density-High (RPS) Thermal Margin/Low Pressure (RPS)
2. Pressurizer Pressure -	Pressurizer Pressure - High (RPS) Thermal Margin/Low Pressure (RPS) Pressurizer Pressure-Low (ESF)
3. Containment Pressure -	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4. Steam Generator Pressure -	Steam Generator Pressure - Low (RPS) Thermal Margin/Low Pressure (RPS) AFAS-1 and AFAS-2 (AFAS) Steam Generator Pressure-Low (ESF)
5. Steam Generator Level -	Steam Generator Level - Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

ACTION 2 - (Continued)

6. Cold Leg Temperature	Variable Power Level – High (RPS) Thermal Margin/Low Pressure (RPS) Local Power Density – High (RPS)
7. Hot Leg Temperature	Variable Power Level – High (RPS) Thermal Margin /Low Pressure (RPS) Local Power Density – High (RPS)

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

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirements, STARTUP and/or POWER OPERATION may continue provided the reactor trip breakers of the inoperable channel are placed in the tripped condition within 1 hour, otherwise, be in at least HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 1 hour, provided the trip breakers of any inoperable channel are in the tripped condition, for surveillance testing per Specification 4.3.1.1.

ACTION 5 - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement restore the inoperable channel to OPERABLE status within 48 hours or open the reactor trip breakers within the next hour.

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* Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

DELETED

DELETED

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, 3*, 4*, 5*
2. Variable Power Level - High				
a. Nuclear Power	S	D(2),M(3),Q(4)	M	1, 2
b. ΔT Power	S	D(5),Q(4)		1
3. Pressurizer Pressure - High	S	R	M	1, 2
4. Thermal Margin/Low Pressure	S	R	M	1, 2
5. Containment Pressure - High	S	R	M	1, 2
6. Steam Generator Pressure - Low	S	R	M	1, 2
7. Steam Generator Pressure Difference - High	S	R	M	1, 2
8. Steam Generator Level - Low	S	R	M	1, 2
9. Local Power Density - High	S	R	M	1
10. Loss of Component Cooling Water to Reactor Coolant Pumps	N.A.	N.A.	M	N.A.
11. Reactor Protection System Logic	N.A.	N.A.	M(7)	1, 2, 3*, 4*, 5*

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
12. Reactor Trip Breakers	N.A.	N.A.	S/U(1),M,R(6)	1, 2, 3*, 4*, 5*
13. Wide Range Logarithmic Neutron Flux Monitor	S	R	S/U(1),R	1, 2, 3, 4, 5
14. Reactor Coolant Flow-Low	S	R	M	1, 2
15. Loss of Load (Turbine Hydraulic Fluid Pressure - Low)	S	N.A.	M	1

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - Only if the reactor trip breakers are in the closed position and the CEA drive system is 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.

(2) - Heat balance only (CHANNEL FUNCTIONAL TEST not included), above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Power - ΔT Power". 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, recalibrate the excore detectors which monitor the AXIAL SHAPE INDEX by using the incore detectors or restrict THERMAL POWER during subsequent operations to $\leq$ 90% of the maximum allowed THERMAL POWER level with the existing reactor coolant pump combination.

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

(5) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

(6) - At least once per 18 months and following maintenance or adjustment of the reactor trip breakers, the CHANNEL FUNCTIONAL TEST shall include verification of the independent OPERABILITY of the undervoltage and shunt trips.

(7) - The fuse circuitry in the matrix fault protection circuitry shall be determined to be OPERABLE by testing with the installed test circuitry.

INSTRUMENTATION

3/4.3.2 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The Engineered Safety Features Actuation System (ESFAS) instrumentation channels and 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.

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 FEATURES 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	12
b. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
c. Pressurizer Pressure – Low	4	2	3	1, 2, 3(a)	13*, 14
d. Automatic Actuation – Logic	2	1	2	1, 2, 3, 4	12
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Containment Pressure – High-High	4	2	3	1(b), 2(b), 3(b)	18a*, 18b*, 18c
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12
3. CONTAINMENT ISOLATION (CIAS)					
a. Manual CIAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Safety Injection (SIAS)	See Functional Unit 1 for all Safety Injection Initiating Functions and Requirements				
c. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
d. Containment Radiation – High	4	2	3	1, 2, 3	13*, 14
e. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3	16
b. Steam Generator Pressure – Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	13*, 14
c. Containment Pressure – High	4	2	3	1, 2, 3	13*, 14
d. Automatic Actuation Logic	2	1	2	1, 2, 3	12
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	12
b. Refueling Water Storage Tank - Low	4	2	3	1, 2, 3	19
c. Automatic Actuation Logic	2	1	2	1, 2, 3	12

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
6. LOSS OF POWER (LOV)					
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	2/Bus	1/Bus	1, 2, 3	17
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	3/Bus	2/Bus	2/Bus	1, 2, 3	17
7. AUXILIARY FEEDWATER (AFAS)					
a. Manual (Trip Buttons)	4/SG	2/SG	4/SG	1, 2, 3	15
b. Automatic Actuation Logic	4/SG	2/SG	3/SG	1, 2, 3	15
c. SG Level (2A/2B) – Low	4/SG	2/SG	3/SG	1, 2, 3	20a*, 20b*, 20c
8. AUXILIARY FEEDWATER ISOLATION					
a. SG 2A – SG 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	20a*, 20b*, 20c
b. Feedwater Header 2A – 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	20a*, 20c

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is less than 1836 psia; bypass shall be automatically removed when pressurizer pressure is greater than or equal to 1836 psia.

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

(c) Trip function may be bypassed in this MODE below 700 psia; bypass shall be automatically removed at or above 700 psia.

* The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 12 - 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 13 - With the number of channels OPERABLE one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may continue provided the inoperable channel is placed in the bypassed or tripped condition within 1 hour. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6m. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

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 -	Containment Pressure - High (SIAS, CIAS, CSAS) Containment Pressure - High (RPS)
2.	Steam Generator Pressure -	Steam Generator Pressure - Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure-Low (RPS)
3.	Steam Generator Level -	Steam Generator Level - Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4.	Pressurizer Pressure -	Pressurizer Pressure - High (RPS) Pressurizer Pressure - Low (SIAS) Thermal Margin/Low Pressure (RPS)

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 14 - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, STARTUP and/or POWER OPERATION may continue 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 -	Containment Pressure - High (SIAS, CIAS, CSAS) Containment Pressure - High (RPS)
2.	Steam Generator Pressure -	Steam Generator Pressure - Low (MSIS) AFAS-1 and AFAS-2 (AFAS) Thermal Margin/Low Pressure (RPS) Steam Generator Pressure-Low (RPS)
3.	Steam Generator Level -	Steam Generator Level-Low (RPS) If SG-2A, then AFAS-1 (AFAS) If SG-2B, then AFAS-2 (AFAS)
4.	Pressurizer Pressure -	Pressurizer Pressure - High (RPS) Pressurizer Pressure - Low (SIAS) Thermal Margin/Low Pressure (RPS)

ACTION 15 - 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 16 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or declare the associated valve inoperable and take the ACTION required by Specification 3.7.1.5.

ACTION 17 - 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 place the inoperable channel in the tripped condition and verify that the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.2.1.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 18 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then place the inoperable channel in the tripped condition.

b. With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel has been placed in the tripped condition within 1 hour. Restore one of 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 19 - With the number of OPERABLE Channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. Within 1 hour the inoperable channel is placed in either the bypassed or tripped condition. If OPERABILITY can not be restored within 48 hours, be in at least HOT STANDBY within 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 20 - With the number of channels OPERABLE one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. If the inoperable channel can not be restored to OPERABLE status within 48 hours, then both AFAS-1 and AFAS-2 in the inoperable channel shall be placed in the bypassed condition. If the inoperable channel is bypassed, the desirability of maintaining this channel in the bypassed condition shall be reviewed in accordance with Specification 6.5.1.6m. The channel shall be returned to OPERABLE status no later than during the next COLD SHUTDOWN.

b With a channel process measurement circuit that affects multiple functional units inoperable or in test, bypass or trip all associated functional units as listed in ACTION 13.

c. With the number of channels OPERABLE one less than the Minimum Channels OPERABLE, operation may proceed provided one of the inoperable channels has been bypassed and the other inoperable channel placed in the tripped condition within 1 hour. Restore one of 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.

TABLE 3.3-4

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 3.5 psig	≤ 3.6 psig
c. Pressurizer Pressure - Low	≥ 1736 psia	≥ 1728 psia
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	≤ 5.40 psig	≤ 5.50 psig
c. Automatic Actuation Logic	Not Applicable	Not Applicable
3. CONTAINMENT ISOLATION (CIAS)		
a. Manual CIAS (Trip Buttons)	Not Applicable	Not Applicable
b. Safety Injection (SIAS)	Not Applicable	Not Applicable
c. Containment Pressure - High	≤ 3.5 psig	≤ 3.6 psig
d. Containment Radiation - High	≤ 10 R/hr	≤ 10 R/hr
e. Automatic Actuation Logic	Not Applicable	Not Applicable
4. MAIN STEAM LINE ISOLATION		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 600 psia	≥ 567 psia
c. Containment Pressure - High	≤ 3.5 psig	≤ 3.6 psig
d. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Storage Tank - Low	5.67 feet above tank bottom	4.62 feet to 6.24 feet above tank bottom
c. Automatic Actuation Logic	Not Applicable	Not Applicable
6. LOSS OF POWER		
a. (1) 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	≥ 3120 volts	≥ 3120 volts
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	≥ 360 volts	≥ 360 volts
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	≥ 3848 volts with < 10-second time delay	≥ 3848 volts with < 10-second time delay
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	≥ 432 volts	≥ 432 volts
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. SG 2A & 2B Level Low	≥ 19.0%	≥ 18.0%
8. AUXILIARY FEEDWATER ISOLATION		
a. Steam Generator ΔP - High	≤ 275 psid	89.2 to 281 psid
b. Feedwater Header ΔP - High	≤ 150.0 psid	56.0 to 157.5 psid

DELETED

DELETED

DELETED

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION 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	1, 2, 3, 4
b. Containment Pressure - High	S	R	M	1, 2, 3
c. Pressurizer Pressure - Low	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
2. CONTAINMENT SPRAY (CSAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Containment Pressure - High-High	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
3. CONTAINMENT ISOLATION (CIAS)				
a. Manual CIAS (Trip Buttons)	N.A.	N.A.	R	1, 2, 3, 4
b. Safety Injection SIAS	N.A.	N.A.	R	1, 2, 3, 4
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Containment Radiation - High	S	R	M	1, 2, 3
e. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
4. MAIN STEAM LINE ISOLATION				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Pressure - Low	S	R	M	1, 2, 3
c. Containment Pressure - High	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3, 4
5. CONTAINMENT SUMP RECIRCULATION (RAS)				
a. Manual RAS (Trip Buttons)	N.A.	N.A.	R	N.A.
b. Refueling Water Storage Tank - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1), R(3)	1, 2, 3

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
6. LOSS OF POWER (LOV)				
a. 4.16 kV and 480 V Emergency Bus Undervoltage (Loss of Voltage)	S	R	R	1, 2, 3, 4
b.. 4.16 kV and 480 V Emergency Bus Undervoltage (Degraded Voltage)	S	R	R	1, 2, 3, 4
7. AUXILIARY FEEDWATER (AFAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	1, 2, 3
b. SG Level (A/B) - Low	S	R	M	1, 2, 3
c. Automatic Actuation Logic	N.A.	N.A.	M(1), SA(2)	1, 2, 3
8. AUXILIARY FEEDWATER ISOLATION				
a. SG Level (A/B) - Low and SG Differential Pressure (B to A/A to B) - High	N.A.	R	M	1, 2, 3
b. SG Level (A/B) - Low and Feedwater Header Differential Pressure (B to A/A to B) - High	N.A.	R	M	1, 2, 3

TABLE NOTATION

(1) Testing of Automatic Actuation Logic shall include energization/de-energization of each initiation relay (solid-state component) and verification of the OPERABILITY of each initiation relay (solid-state component).

(2) An actuation relay test shall be performed which shall include the energization/de-energization of each actuation relay and verification of the OPERABILITY of each actuation relay.

(3) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay and verification of the OPERABILITY of each subgroup relay. Testing of the ESFAS subgroup relays shall be performed on a STAGGERED TEST BASIS at subintervals of 6 months, such that each subgroup relay is tested at least once per 18 months.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-6
RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area					
i. Criticality and Ventilation System Isolation Monitor	4	*	≤ 20 mR/hr	10^{-1} - 10^{4} mR/hr	22
b. Containment Isolation	3	6	≤ 90 mR/hr	1 - 10^{7} mR/hr	25
c. Control Room Isolation	1 per intake	ALL MODES	≤ 2x background	10^{-7} - 10^{-2} µCi/cc	26
d. Containment Area - Hi Range	1	1, 2, 3 & 4	Not Applicable	1 - 10^{7} R/hr	27
2. PROCESS MONITORS					
a. Fuel Storage Pool Area Ventilation System					
i. Gaseous Activity	1	**	***	10^{-7} - 10^{-2} µCi/cc	24
ii. Particulate Activity	1	**	***	1 - 10^{6} cpm	24
b. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{-7} - 10^{-2} µCi/cc	23
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	1 - 10^{6} cpm	23

*With fuel in the storage pool or building.

**With irradiated fuel in the storage pool or whenever there is fuel movement within the pool or crane operation with loads over the storage pool.

***The Alarm/Trip Setpoints are determined and set in accordance with the requirements of the Offsite Dose Calculation Manual.

TABLE 3.3-6 (Continued)

RADIATION MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ALARM/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
PROCESS MONITORS (Continued)					
c. Noble Gas Effluent Monitors					
i. Reactor Auxiliary Building Exhaust System (Plant Vent Low Range Monitor)	1	1, 2, 3 & 4	***	10^{-7} - 10^{-2} µCi/cc	27
ii. Reactor Auxiliary Building Exhaust System (Plant Vent High Range Monitor)	1	1, 2, 3 & 4	***	10^{-2} - 10^{5} µCi/cc	27
iii. Steam Generator Blowdown Treatment Facility Building Exhaust System	1	1, 2, 3 & 4	***	10^{-7} - 10^{-2} µCi/cc	27
iv. Steam Safety Valve Discharge#	1/steam header	1, 2, 3 & 4	***	10^{-1} - 10^{3} µCi/cc	27
v. Atmospheric Steam Dump Valve Discharge#	1/steam header	1, 2, 3 & 4	***	10^{-1} - 10^{3} µCi/cc	27
vi. ECCS Exhaust	1/train	1, 2, 3 & 4	***	10^{-7} - 10^{5} µCi/cc	27

*** The Alarm/Trip Setpoints are determined and set in accordance with the requirements of the Offsite Dose Calculation Manual.
# The steam safety valve discharge monitor and the atmospheric steam dump valve discharge monitor are the same monitor.

TABLE 3.3-6 (Continued)

ACTION STATEMENTS

ACTION 22 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 23 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.4.6.1.

ACTION 24 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, suspend all operations involving movement of fuel within the spent fuel storage pool and crane operations with loads over the spent fuel storage pool.

ACTION 25 - 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 26 - 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 27 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

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

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

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

* With fuel in the storage pool or building

** With irradiated fuel in the storage pool

TABLE 4.3-3 (Continued)

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
PROCESS MONITORS (Continued)				
c. Noble Gas Effluent Monitors				
i. Reactor Auxiliary Building Exhaust System (Plant Vent Low Range Monitor)	S	R	M	1, 2, 3 & 4
ii. Reactor Auxiliary Building Exhaust System (Plant Vent High Range Monitor)	S	R	M	1, 2, 3, & 4
iii. Steam Generator Blowdown Treatment Building Exhaust System	S	R	M	1, 2, 3 & 4
iv. Steam Safety Valve Discharge#	S	R	M	1, 2, 3 & 4
v. Atmospheric Steam Dump Valve Discharge#	S	R	M	1, 2, 3 & 4
vi. ECCS Exhaust	S	R	M	1, 2, 3 & 4

The steam safety valve discharge monitor and the atmospheric steam dump valve discharge monitor are the same monitor.

PAGE 3/4 3-31 (ORIGINAL) HAS BEEN DELETED FROM THE TECHNICAL SPECIFICATIONS.

THE NEXT PAGE IS 3/4 3-32.

Pages 3/4 3-33 through 3/4 3-37 have been DELETED.

The next page is 3/4 3-38.

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.5 The remote shutdown system transfer switches, controls and instrumentation channels shown in Table 3.3-9 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the number of OPERABLE remote shutdown channels less than the Required Number of Channels shown in Table 3.3-9, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE remote shutdown channels less than the Minimum Channels OPERABLE requirements of 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.

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.5.1 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.

4.3.3.5.2 Each remote shutdown system instrumentation transfer switch and control circuit shall be demonstrated OPERABLE by verifying its capability to perform its intended function(s) at least once per 18 months.

TABLE 3.3-9

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	CHANNELS RANGE	REQUIRED OF NUMBER CHANNELS	MINIMUM CHANNELS OPERABLE
1. Power Range Neutron Flux	Hot Shutdown Panel	2×10^{-8}% - 200%	2	1
2. Reactor Trip Breaker Indication	Reactor Trip Switch Gear (RB)	OPEN-CLOSE	1/trip breaker	1/trip breaker
3. Reactor Coolant Temperature - T_{Cold}	Hot Shutdown Panel	0° - 600°F	2	1
4. Pressurizer Pressure	Hot Shutdown Panel	0 - 3000 psia	2	1
5. Pressurizer Level	Hot Shutdown Panel	0 - 100% level	2	1
6. Steam Generator Pressure	Hot Shutdown Panel	0 - 1200 psia	1/steam generator	1/steam generator
7. Steam Generator Level	Hot Shutdown Panel	0 - 100% level	2/steam generator	1/steam generator
8. Shutdown Cooling Flow Rate	Hot Shutdown Panel	0 - 5000 gpm	2	1
9. Shutdown Cooling Temperature	Hot Shutdown Panel	0° - 350°F	2	1
10. Diesel Generator Voltage	Hot Shutdown Panel	0 - 5250 V	1/diesel generator	1/diesel generator
11. Diesel Generator Power	Hot Shutdown Panel	0 - 5000 kW	1/diesel generator	1/diesel generator
12. Atmospheric Dump Valve Pressure	Hot Shutdown Panel	0 - 1200 psig	1/steam generator	1/steam generator
13. Charging Flow/Pressure	Hot Shutdown Panel	0 - 150 gpm/ 0 - 3000 psia	2	1

CONTROLS/ISOLATE SWITCHES

1. Atmospheric Stm Dump Controllers	Hot Shutdown Panel/RAB431	N.A.	2/steam generator	1/steam generator
2. Aux. Spray Valves	Hot Shutdown Panel/RAB431	N.A.	2	1
3. Charging Pump Controls	Hot Shutdown Panel/RAB431	N.A.	3	2
4. Letdown Isol Valve	Hot Shutdown Panel/RAB431	N.A.	3	2
5. AFW Pump/Valve Controls	Hot Shutdown Panel/RAB431	N.A.	3	2
6. AFW Pump Steam Inlet Valve	Hot Shutdown Panel/RAB431	N.A.	2	1
7. Pzr Heater Controls	Hot Shutdown Panel/RAB431	N.A.	6	3

TABLE 4.3-6

REMOTE SHUTDOWN SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Power Range Neutron Flux	M	Q
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Temperature- T_{Cold}	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Level	M	R
7. Steam Generator Pressure	M	R
8. Shutdown Cooling Flow Rate	M	R
9. Shutdown Cooling Temperature	M	R
10. Diesel Generator Voltage	M	R
11. Diesel Generator Power	M	R
12. Atmospheric Dump Valve Pressure	M	R
13. Charging Flow/Pressure	M	R

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a.* With the number of OPERABLE accident monitoring channels less than the 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.** With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-10, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

d.** With the number of OPERABLE 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 if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and
2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status, and
3. Restore the Channel to OPERABLE status at the next scheduled refueling.

e. The provisions of Specification 3.0.4 are not applicable.

* Action statements do not apply to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

**Action statements apply only to Reactor Vessel Level Monitoring System, Containment Sump Water Level (narrow range) and Containment Sump Water Level (wide range) instruments.

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Containment Pressure	2	1
2. Reactor Coolant Outlet Temperature – T_{Hot} (Wide Range)	2	1
3. Reactor Coolant Inlet Temperature – T_{Cold} (Wide Range)	2	1
4. Reactor Coolant Pressure – Wide Range	2	1
5. Pressurizer Water Level	2	1
6. Steam Generator Pressure	2/steam generator	1/steam generator
7. Steam Generator Water Level – Narrow Range	1/steam generator	1/steam generator
8. Steam Generator Water Level – Wide Range	1/steam generator*	1/steam generator*
9. Refueling Water Storage Tank Water Level	2	1
10. Auxiliary Feedwater Flow Rate (Each pump)	1/pump*	1/pump*
11. Reactor Cooling System Subcooling Margin Monitor	2	1
12. PORV Position/Flow Indicator	2/valve***	1/valve**
13. PORV Block Valve Position Indicator	1/valve**	1/valve**
14. Safety Valve Position/Flow Indicator	1/valve***	1/valve***
15. Containment Sump Water Level (Narrow Range)	1****	1****
16. Containment Water Level (Wide Range)	2	1
17. Incore Thermocouples	4/core quadrant	2/core quadrant
18. Reactor Vessel Level Monitoring System	2*****	1*****

* These corresponding instruments may be substituted for each other.

** Not required if the PORV block valve is shut and power is removed from the operator.

*** If not available, monitor the quench tank pressure, level and temperature, and each safety valve/PORV discharge piping temperature at least once every 12 hours.

**** The non-safety grade containment sump water level instrument may be substituted.

***** <u>Definition of OPERABLE</u>: A channel consists of eight (8) sensors in a probe of which four (4) sensors must be OPERABLE.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Containment Pressure	M	R
2. Reactor Coolant Outlet Temperature – T_{Hot} (Wide Range)	M	R
3. Reactor Coolant Inlet Temperature – T_{Cold} (Wide Range)	M	R
4. Reactor Coolant Pressure – Wide Range	M	R
5. Pressurizer Water Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Water Level – Narrow Range	M	R
8. Steam Generator Water Level – Wide Range	M	R
9. Refueling Water Storage Tank Water Level	M	R
10. Auxiliary Feedwater Flow Rate (Each pump)	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. PORV Position/Flow Indicator	M	R
13. PORV Block Valve Position Indicator	M	R
14. Safety Valve Position/Flow Indicator	M	R
15. Containment Sump Water Level (Narrow Range)	M	R
16. Containment Water Level (Wide Range)	M	R
17. Incore Thermocouples	M	R
18. Reactor Vessel Level Monitoring System	M	R

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

STARTUP AND POWER OPERATION

LIMITING CONDITION FOR OPERATION

3.4.1.1 Both Reactor Coolant loops and both Reactor Coolant pumps in each loop shall be in operation.

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.

REACTOR COOLANT SYSTEM

HOT STANDBY

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant Loop 2A and its associated steam generator and at least one associated Reactor Coolant pump.

b. Reactor Coolant Loop 2B and its associated steam generator and at least one associated Reactor Coolant pump.

APPLICABILITY: MODE 3

ACTION:

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

b. With no Reactor Coolant loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.1 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 ≥ 10% indicated narrow range level at least once per 12 hours.

---

* All Reactor Coolant pumps may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.1 and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.1.3 At least two of the loop(s)/train(s) listed below shall be OPERABLE and at least one Reactor Coolant and/or shutdown cooling loops shall be in operation.*

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

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

c. Shutdown Cooling Train 2A,

d. Shutdown Cooling Train 2B.

APPLICABILITY: MODE 4.

ACTION:

a. With less than the above required Reactor Coolant and/or shutdown cooling loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible; if the remaining OPERABLE loop is a shutdown cooling loop, be in COLD SHUTDOWN within 30 hours.

b. With no Reactor Coolant or shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specifications 3.1.1.1 and immediately initiate corrective action to return the required coolant loop to operation.

---

* All Reactor Coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.1 and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

** A Reactor Coolant pump shall not be started with two idle loops and one or more of the Reactor Coolant System cold leg temperatures less than or equal to that specified in Table 3.4-3 unless the secondary water temperature of each steam generator is less than 40°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

HOT SHUTDOWN

SURVEILLANCE REQUIREMENTS

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

4.4.1.3.2 The required steam generator(s) shall be determined OPERABLE by verifying the secondary side water level to be $\geq$ 10% indicated narrow range level at least once per 12 hours.

4.4.1.3.3 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.

COLD SHUTDOWN – LOOPS FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.1 At least one shutdown cooling loop shall be OPERABLE and in operation*, and either:

a. One additional shutdown cooling loop shall be OPERABLE#, or

b. The secondary side water level of at least two steam generators shall be greater than 10% indicated narrow range level.

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

ACTION:

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

b. With no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and immediately initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

---

* The shutdown cooling pump may be de-energized for up to 1 hour provided 1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and 2) core outlet temperature is maintained at least 10°F below saturation temperature.

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

A Reactor Coolant pump shall not be started with two idle loops unless the secondary water temperature of each steam generator is less than 40°F above each of the Reactor Coolant System cold leg temperatures.

REACTOR COOLANT SYSTEM

COLD SHUTDOWN – LOOPS NOT FILLED

LIMITING CONDITION FOR OPERATION

3.4.1.4.2 Two shutdown cooling loops shall be OPERABLE# and at least one shutdown cooling loop shall be in operation.*

APPLICABILITY: MODE 5 with reactor coolant loops not filled.

ACTION:

a. With less than the above required loops OPERABLE, within 1 hour initiate corrective action to return the required loops to OPERABLE status as soon as possible.

b. With no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and within 1 hour initiate corrective action to return the required shutdown cooling loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

* The shutdown cooling pump may be deenergized for up to 1 hour provided (1) no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.1.1.2 and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

DELETED

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with a lift setting of $\geq$ 2435.3 psig and $\leq$ 2535.3 psig.*

APPLICABILITY: MODES 1, 2, 3, and 4 with all RCS cold leg temperatures > 230°F.

ACTION:

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

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures at $\leq$ 230°F within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 Verify each pressurizer code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within +/- 1% of 2500 psia.

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

REACTOR COOLANT SYSTEM

3/4.4.3 PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.3 The pressurizer shall be OPERABLE with a minimum water level of greater than or equal to 27% indicated level and a maximum water level of less than or equal to 68% indicated level and at least two groups of pressurizer heaters capable of being powered from 1E buses each having a nominal capacity of at least 150 kW.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.4.3.2 The capacity of each of the above required groups of pressurizer heaters shall be verified to be at least 150 kW at least once per 92 days.

4.4.3.3 The emergency power supply for the pressurizer heaters shall be demonstrated OPERABLE at least once per 18 months by verifying that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power:

a. the pressurizer heaters are automatically shed from the emergency power sources, and

b. the pressurizer heaters can be reconnected to their respective buses manually from the control room after resetting of the ESFAS test signal.

3/4.4.4 PORV BLOCK VALVES

LIMITING CONDITION FOR OPERATION

3.4.4 Each Power Operated Relief Valve (PORV) Block valve shall be OPERABLE. No more than one block valve shall be open at any one time.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

b. With both block valves open, close one block valve within 1 hour, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

3/4.4.5 STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

4.4.5.2 Steam Generator Tube Sample Selection and Inspection - The steam generator tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 4.4-2. The inservice inspection of steam generator tubes shall be performed at the frequencies specified in Specification 4.4.5.3 and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 4.4.5.4. The tubes selected for each inservice inpection 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:

1. All nonplugged tubes that previously had detectable wall penetrations (greater than 20%).
2. Tubes in those areas where experience has indicated potential problems.
3. A tube inspection (pursuant to Specification 4.4.5.4.a.8) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection.

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

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes wtih 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 of 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.

Category	Inspection Results
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

Note: (1) In all inspections, previously degraded tubes must exhibit significant (greater than 10%) further wall penetrations to be included in the above percentage calculations.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

   3. A loss-of-coolant accident requiring actuation of the Engineered Safety Features.

   4. A main steam line or feedwater line break.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.5.4 Acceptance Criteria

a. As used in this Specification

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

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

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

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

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

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

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

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

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

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

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

4.4.5.5 Reports

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

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

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

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

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

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.	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.	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

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following RCS leakage detection systems shall be OPERABLE:

a. The reactor cavity sump inlet flow monitoring system; and

b. One containment atmosphere radioactivity monitor (gaseous or particulate).

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

ACTION:

a. With the required reactor cavity sump inlet flow monitoring system inoperable, perform a RCS water inventory balance at least once per 24 hours and restore the sump inlet flow monitoring system to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the required radioactivity monitor inoperable, analyze grab samples of the containment atmosphere or perform a RCS water inventory balance at least once per 24 hours, and restore the required radioactivity monitor to OPERABLE status within 30 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With all required monitors inoperable, enter LCO 3.0.3 immediately.

d. The provisions of Specification 3.0.4 are not applicable if at least one of the required monitors is OPERABLE.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The RCS leakage detection instruments shall be demonstrated OPERABLE by:

a. Performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION of the required containment atmosphere radioactivity monitor at the frequencies specified in Table 4.3-3.

b. Performance of the CHANNEL CALIBRATION of the required reactor cavity sump inlet flow monitoring system at least once per 18 months.

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 gpm UNIDENTIFIED LEAKAGE,

c. 1 gpm total primary-to-secondary leakage through all steam generators and 720 gallons per day through any one steam generator,

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

e. 1 gpm leakage (except as noted in Table 3.4-1) at a Reactor Coolant System pressure of 2235 ± 20 psig from any Reactor Coolant System Pressure Isolation Valve specified in Table 3.4-1.

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

ACTION:

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

b. With any Reactor Coolant System leakage greater than any one of the 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.

d. With RCS leakage alarmed and confirmed in a flow path with no flow indication, commence an RCS water inventory balance within 1 hour to determine the leak rate.

SURVEILLANCE REQUIREMENTS

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

4.4.6.2.2 Each Reactor Coolant System Pressure Isolation Valve check valve specified in Table 3.4-1 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 7 days 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. Following valve actuation due to automatic or manual action or flow through the valve:

1. Within 24 hours by verifying valve closure, and

2. Within 31 days by verifying leakage rate.

4.4.6.2.3 Each Reactor Coolant System Pressure Isolation Valve motor-operated valve specified in Table 3.4-1 shall be demonstrated OPERABLE by verifying leakage to be within its limit;

a. At least once per 18 months, and

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

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

TABLE 3.4-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

Check Valve No.		Motor-Operated Valve No.
V3217	V3525	V3480
V3227	V3524	V3481
V3237	V3527	V3652
V3247	V3526	V3651
V3259		
V3258		
V3260		
V3261		
V3215		
V3225		
V3235		
V3245		

NOTES

(a) Maximum Allowable Leakage (each valve):

1. Except as noted below leakage rates greater than 1.0 gpm are unacceptable.

2. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between previous measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. For motor-operated valves (MOVs) only, leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are unacceptable.

(b) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

(c) Minimum test differential pressure shall not be less than 200 psid.

3/4.4.7 CHEMISTRY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: All MODES

ACTION:

MODES 1, 2, 3, and 4:

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

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

MODES 5 and 6:

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.4-2

REACTOR COOLANT SYSTEM

CHEMISTRY

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

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

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	MINIMUM SAMPLING FREQUENCIES	MAXIMUM TIME BETWEEN SAMPLES
DISSOLVED OXYGEN	3 times per 7 days*	72 hours
CHLORIDE	3 times per 7 days	72 hours
FLUORIDE	3 times per 7 days	72 hours

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

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

MODES 1, 2 and 3*:

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

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

MODES 1, 2, 3, 4 and 5.

With the specific activity of the primary coolant greater than 1 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.

SURVEILLANCE REQUIREMENTS

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

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

DELETED

TABLE 4.4-4

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

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

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

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

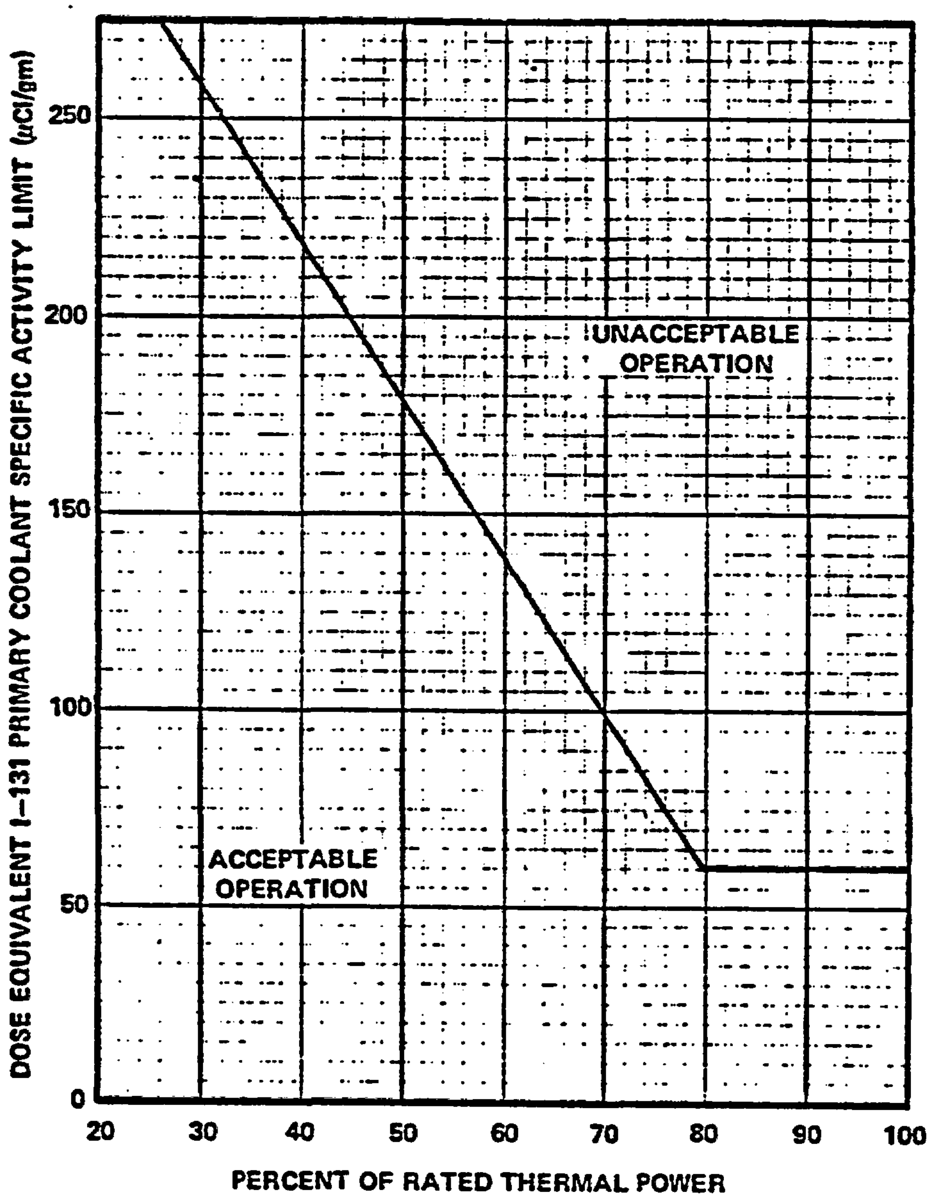


FIGURE 3.4-1

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3 and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing.

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} to less than 200°F within the next 30 hours in accordance with Figures 3.4-3 and 3.4-4.

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS (Continued)

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR 50 Appendix H. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.

FIGURE 3.4-2
ST. LUCIE-2 P/T LIMITS, 21.7 EFPY
HEATUP AND CORE CRITICAL

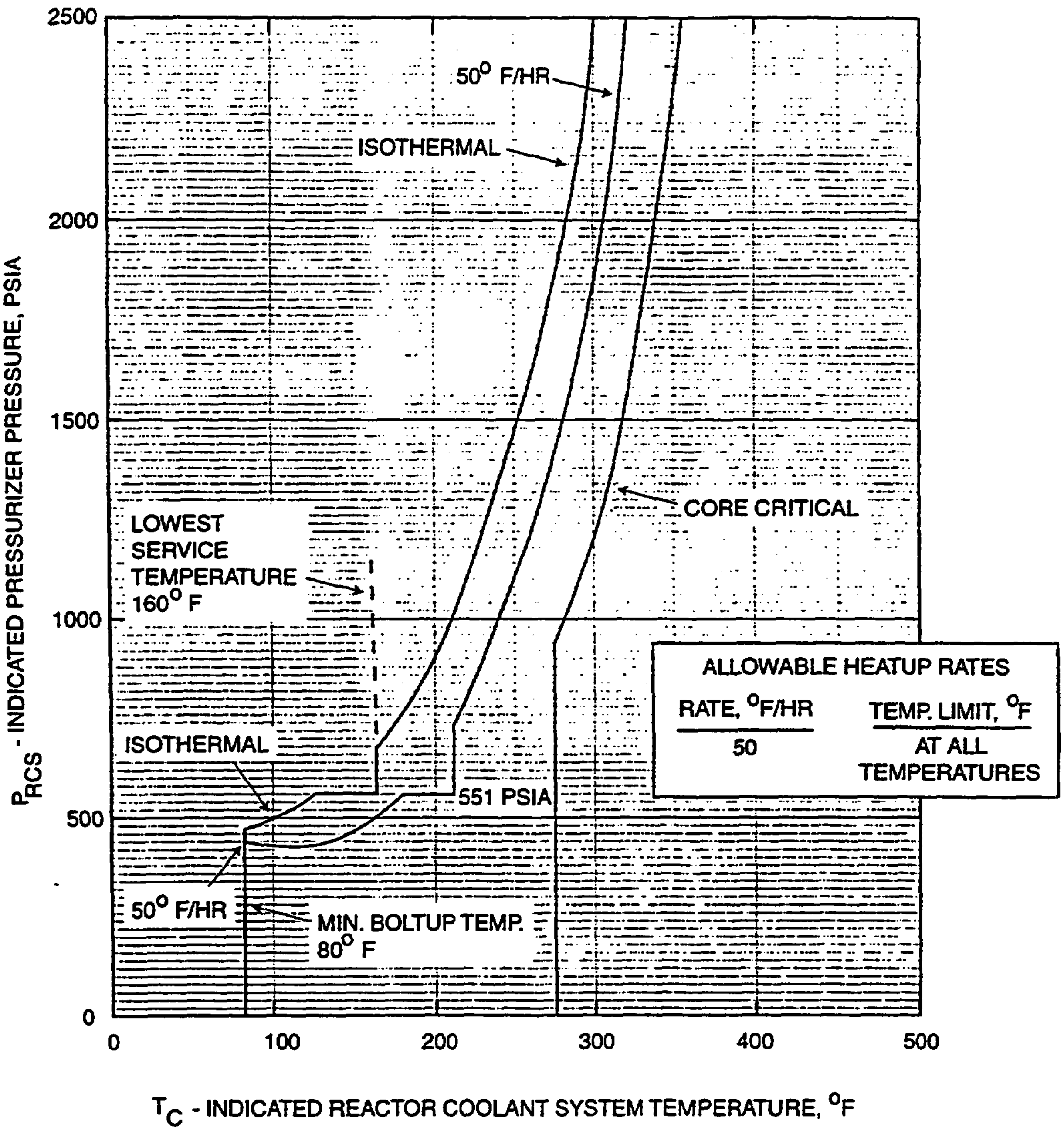


FIGURE 3.4-3
ST. LUCIE-2 P/T LIMITS, 21.7 EFPY
COOLDOWN AND INSERVICE-TEST

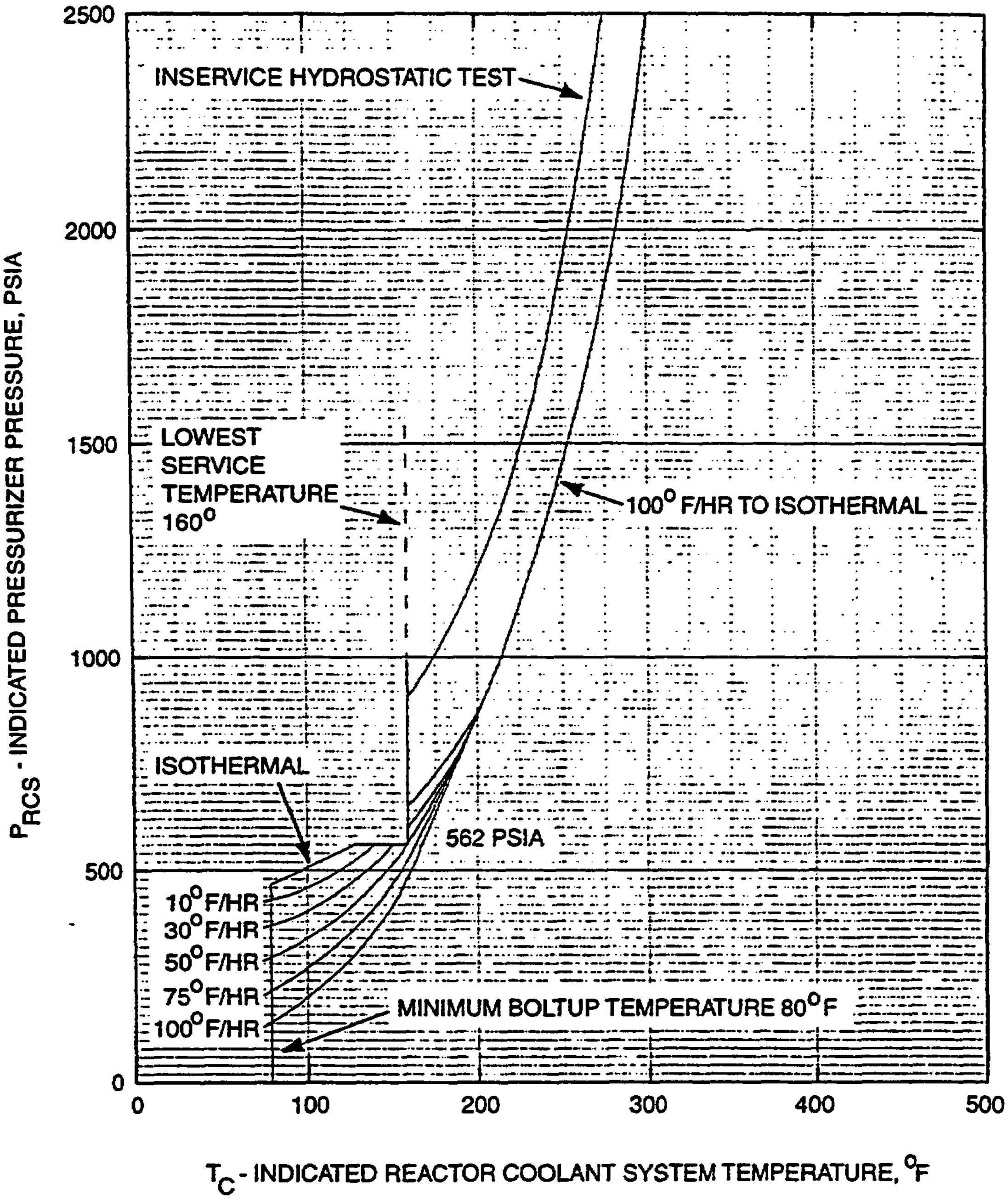
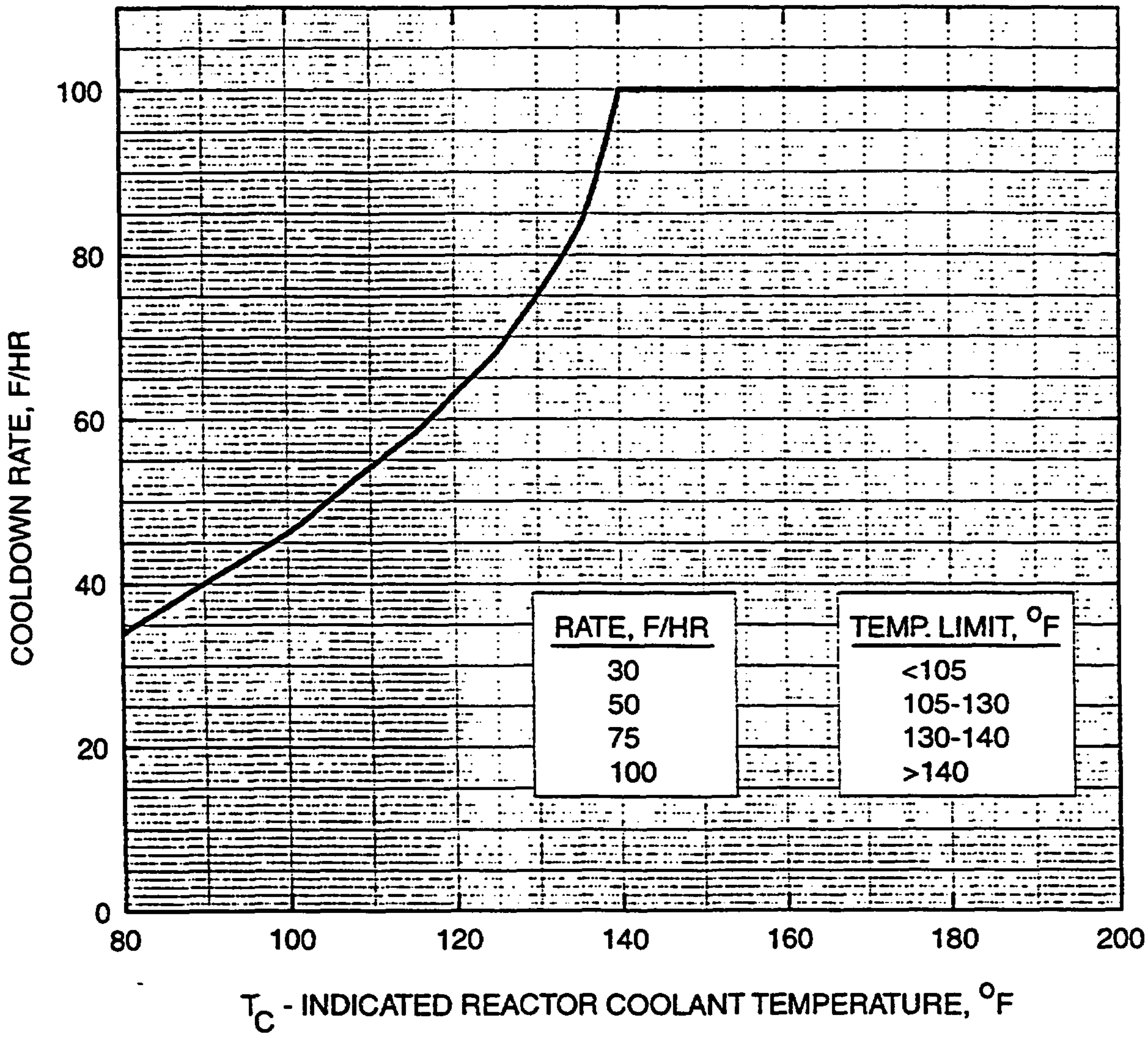


FIGURE 3.4-4
ST. LUCIE-2 P/T LIMITS, 21.7 EFPY
MAXIMUM ALLOWABLE COOLDOWN RATES



RATE, F/HR	TEMP. LIMIT, °F
30	<105
50	105-130
75	130-140
100	>140

NOTE: A MAXIMUM COOLDOWN RATE OF 100 F/HR IS ALLOWED AT ANY TEMPERATURE ABOVE 140°F.

DELETED

PRESSURIZER HEATUP/COOLDOWN LIMITS

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

b. A maximum cooldown of 200°F in any 1-hour period.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown.

REACTOR COOLANT SYSTEM

OVERPRESSURE PROTECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 Unless the RCS is depressurized and vented by at least 3.58 square inches, at least one of the following overpressure protection systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of less than or equal to 470 psia and with their associated block valves open. These valves may only be used to satisfy low temperature overpressure protection (LTOP) when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

b. Two shutdown cooling relief valves (SDCRVs) with a lift setting of less than or equal to 350 psia.

c. One PORV with a lift setting of less than or equal to 470 psia and with its associated block valve open in conjunction with the use of one SDCRV with a lift setting of less than or equal to 350 psia. This combination may only be used to satisfy LTOP when the RCS cold leg temperature is greater than the temperature listed in Table 3.4-4.

APPLICABILITY: MODES 4#, 5 and 6.

ACTION:

a. With either a PORV or an SDCRV being used for LTOP inoperable, restore at least two overpressure protection devices to OPERABLE status within 7 days or:

   1. Depressurize and vent the RCS with a minimum vent area of 3.58 square inches within the next 8 hours; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3 within the next 8 hours.

b. With none of the overpressure protection devices being used for LTOP OPERABLE, within the next eight hours either:

   1. Restore at least one overpressure protection device to OPERABLE status or vent the RCS; OR

   2. Be at a temperature above the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

#With cold leg temperature within the LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE of Table 3.4-3.

LIMITING CONDITION FOR OPERATION

ACTION (Continued):

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

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. In addition to the requirements of the Inservice Testing Program, operating the PORV through one complete cycle of full travel at least once per 18 months.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

d. Verifying the PORV isolation valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

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

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

TABLE 3.4-3

LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

Operating Period, EFPY	Cold Leg Temperature, F° During Heatup	Cold Leg Temperature, F° During Cooldown
$\leq$ 21.7	$\leq$ 247	$\leq$ 230

TABLE 3.4-4

MINIMUM COLD LEG TEMPERATURE FOR PORV USE FOR LTOP

Operating Period EFPY	T_{cold}, F° During Heatup	T_{cold}, F° During Cooldown
$\leq$ 21.7	165	165

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

LIMITING CONDITION FOR OPERATION

3.4.10 At least one Reactor Coolant System vent path consisting of two vent valves and one block valve powered from emergency buses shall be OPERABLE and closed at each of the following locations:

a. Pressurizer steam space, and

b. Reactor vessel head.

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

ACTION:

a. With one of the above Reactor Coolant System vent paths inoperable, STARTUP and/or POWER OPERATION may continue provided the inoperable vent path is maintained closed with power removed from the valve actuator of all the vent valves and block valves in the inoperable vent path; restore the inoperable vent path to OPERABLE status within 30 days, or, be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both Reactor Coolant System vent paths inoperable, maintain the inoperable vent paths closed with power removed from the valve actuators of all the vent valves and block valves in the inoperable vent paths, and restore at least one of the vent paths to OPERABLE status within 72 hours or be in HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.10.1 Each Reactor Coolant System vent path shall be demonstrated OPERABLE at least once per 18 months by:

1. Verifying all manual isolation valves in each vent path are locked in the open position.

2. Cycling each vent valve through at least one complete cycle of full travel from the control room.

3. Verifying flow through the Reactor Coolant System vent paths during venting.

3/4.4.11 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.11 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with Specification 4.4.11.

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

4.4.11 In addition to the requirements of Specification 4.0.5, each Reactor Coolant Pump flywheel shall be inspected per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3/4.5.1 SAFETY INJECTION TANKS (SIT)

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System safety injection tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume of between 1420 and 1556 cubic feet,

c. A boron concentration of between 1720 and 2100 ppm of boron, and

d. A nitrogen cover-pressure of between 500 and 650 psig.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one SIT inoperable due to boron concentration not within limits, or due to an inability to verify the required water volume or nitrogen cover-pressure, restore the inoperable SIT to OPERABLE status within 72 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one SIT inoperable due to reasons other than those stated in ACTION-a, restore the inoperable SIT to OPERABLE status within 24 hours; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 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 that the contained borated water volume and nitrogen cover-pressure in the tanks are within their limits, and

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

---

* With pressurizer pressure greater than or equal to 1750 psia. When pressurizer pressure is less than 1750 psia, at least three safety injection tanks shall be OPERABLE, each with a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 1250 and 1556 cubic feet with a boron concentration of between 1720 and 2100 ppm of boron. With all four safety injection tanks OPERABLE, each tank shall have a minimum pressure of 235 psig and a maximum pressure of 650 psig and a contained water volume of between 833 and 1556 cubic feet with a boron concentration of between 1720 and 2100 ppm of boron.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and once 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. This latter surveillance is not required when the volume increase makeup source is the RWT and the RWT has not been diluted since verifying that the RWT boron concentration is equal to or greater than the safety injection tank boron concentration limit.

c. At least once per 31 days when the RCS pressure is above 700 psia, by verifying that power to the isolation valve operator is disconnected by maintaining the breaker open by administrative controls.

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 515 psia, and

2. Upon receipt of a safety injection test signal.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - OPERATING

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 Recirculation Actuation Signal, and

d. One OPERABLE charging pump.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. 1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem 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.

2. With one ECCS subsystem inoperable for reasons other than condition a.1., 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 1750 psia.

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. V3733 V3734	a. SIT Vent Valves	a. Locked Closed
b. V3735 V3736	b. SIT Vent Valves	b. Locked Closed
c. V3737 V3738 V3739 V3740	c. SIT Vent Valves	c. Locked Closed

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

c. By verifying that the ECCS piping is full of water by venting the accessible piping high points following maintenance, shutdown cooling system operation and/or any other activity which could cause the introduction of air into the system.

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

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

   2. At least once daily of the areas affected within containment by the containment entry and during the final entry when CONTAINMENT INTEGRITY is established.

e. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the shutdown cooling system from the Reactor Coolant System when RCS pressure (actual or simulated) is greater than or equal to 515 psia, and that the interlocks prevent opening the shutdown cooling system isolation valves when RCS pressure (actual or simulated) is greater than or equal to 276 psia.

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.

3. Verifying that a minimum total of 173 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 70.5 $\pm$ 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 $\pm$ 0.1 gallons of 120 $\pm$ 10°F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

f. 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/or 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 upon receipt of an actual or simulated Recirculation Actuation Signal: each low-pressure safety injection pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.

g. By verifying that each of the following pumps develops the specified total developed head on recirculation flow when tested pursuant to the Inservice Testing Program:

1. High-Pressure Safety Injection pumps: greater than or equal to 2854 ft.

2. Low-Pressure Safety Injection pump: greater than or equal to 374 ft.

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

1. During valve stroking operation or following maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE.

2. At least once per 18 months.

HPSI System Valve Number	LPSI System Valve Number
a. HCV 3616/3617	a. HCV 3615
b. HCV 3626/3627	b. HCV 3625
c. HCV 3636/3637	c. HCV 3635
d. HCV 3646/3647	d. HCV 3645
e. V3523/V3540	

i. By performing a flow balance test, during shutdown, following completion of modifications to the ECCS subsystems that alter the subsystem flow characteristics. The test shall measure the individual leg flow rates and pump total developed head to verify the following conditions:

1. HPSI Pump 2A:
The sum of the three lowest cold leg flow rates shall be greater than or equal to 476 gpm with total developed head greater than or equal to 1150 ft but less than or equal to 1290 ft.

2. HPSI Pump 2B:
The sum of the three lowest cold leg flow rates shall be greater than or equal to 484 gpm with total developed head greater than or equal to 910 ft but less than or equal to 1040 ft.

3. With the system operating in hot/cold leg injection mode, the hot leg flow shall be greater than or equal to 317 gpm and within 10% of the cold leg header flow and:

HPSI Pump 2A:
The pump shall be producing total developed head greater than or equal to 1297 ft but less than or equal to 1500 ft.

HPSI Pump 2B:
The pump shall be producing total developed head greater than or equal to 1042 ft but less than 1250 ft.

4. LPSI System - Each Pump:

The flow through each injection leg shall be greater than or equal to 1763 gpm at a total developed head greater than or equal to 298 ft but less than or equal to 337 ft.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.3 ECCS SUBSYSTEMS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.5.3 As a minimum, one ECCS subsystem comprised of the following shall be OPERABLE:

a. One OPERABLE high-pressure safety injection pump, and

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

APPLICABILITY: MODES 3* and 4#.
Footnote # shall remain applicable in MODES 5 and 6 when the Pressurizer manway cover is in place and the reactor vessel head is on.

ACTION:

a. With no ECCS subsystems 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 The ECCS subsystem shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

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* With pressurizer pressure less than 1750 psia.

\# One HPSI shall be rendered inoperable prior to entering MODE 5.

EMERGENCY CORE COOLING SYSTEMS

3/4.5.4 REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained borated water volume of 417,100 gallons,

b. A boron concentration of between 1720 and 2100 ppm of boron, and

c. A solution temperature of between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the refueling water 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 RWT 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 55°F or greater than 100°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

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

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within 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

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations ** not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open on an intermittent basis under administrative control.

b. By verifying that each containment air lock is in compliance with the requirements of Specification 3.6.1.3.

* In MODES 1 and 2, the RCB polar crane shall be rendered inoperable by locking the power supply breaker open.

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

CONTAINMENT SYSTEMS

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited in accordance with the Containment Leakage Rate Testing Program.

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

ACTION:

With the containment leakage rate exceeding the acceptance criteria of the Containment Leakage Rate Testing Program, within 1 hour initiate action to be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore the overall leakage rate to less than that specified by the Containment Leakage Rate Testing Program, 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 required test schedule and shall be determined in conformance with the criteria specified in the Containment Leakage Rate Testing Program.

Pages 3/4 6-4 through 3/4 6-8 have been DELETED.

Page 3/4 6-9 is the next valid page.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate in accordance with the Containment Leakage Rate Testing Program.

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.

* If the inner air lock door is inoperable, passage through the OPERABLE outer air lock door is permitted to effect repairs to the inoperable inner air lock door. No more than one airlock door shall be open at any time.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. By verifying leakage rates and air lock door seals in accordance with the Containment Leakage Rate Testing Program; and

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

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.368 and +0.400 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.

AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

Location

a. TE-07-3A NW RCB Elevation 70'

b. TE-07-3B SW RCB Elevation 70'

*With one temperature detector inoperable, use the air intake temperature detectors of the operating containment fan coolers.

CONTAINMENT SYSTEMS

CONTAINMENT VESSEL STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of the containment vessel not conforming to the above requirements, restore the structural integrity to within the limits prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.6 The structural integrity of the containment vessel shall be determined in accordance with the Containment Leakage Rate Testing Program by a visual inspection of the exposed accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation.

CONTAINMENT SYSTEMS

CONTAINMENT VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.7 Each containment purge supply and exhaust isolation valve shall be OPERABLE and:

a. Each 48-inch containment purge supply and exhaust isolation valve shall be sealed closed.

b. The 8-inch containment purge supply and exhaust isolation valves may be open for purging and/or venting as required for safety related purposes such as:

1. Maintaining containment pressure within the limits of Specification 3.6.1.4.

2. Reducing containment atmosphere airborne radioactivity and/or improving air quality to an acceptable level for containment access.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With a 48-inch containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, close and/or seal close the open valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With an 8-inch containment purge supply and/or exhaust isolation valve(s) open for reasons other than those stated in Specification 3.6.1.7.b, close the open 8-inch valve(s) or isolate the penetration(s) within 4 hours, otherwise be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With a containment purge supply and/or exhaust isolation valve(s) having a measured leakage rate exceeding the limits of Surveillance Requirements 4.6.1.7.3 and/or 4.6.1.7.4, restore the inoperable valve(s) to OPERABLE status within 24 hours, otherwise be in at least HOT STANDY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7.1 Each 48-inch containment purge supply and exhaust isolation valve shall be verified to be sealed-closed at least once per 31 days.

4.6.1.7.2 Documentation shall be reviewed every 18 months to confirm that purging and venting were performed in accordance with Specification 3.6.1.7.b.

4.6.1.7.3 At least once per 6 months on a STAGGERED TEST BASIS each sealed closed 48-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

4.6.1.7.4 At least once per 92 days, each 8-inch containment purge supply and exhaust isolation valve with resilient material seals shall be demonstrated OPERABLE by verifying that the measured leakage rate is less than or equal to 0.05 L_a when pressurized to P_a.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY AND COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.6.2.1 Two containment spray trains and two containment cooling trains shall be OPERABLE.

APPLICABILITY: Containment Spray System: MODES 1, 2, and MODE 3 with Pressurizer Pressure ≥ 1750 psia.

Containment Cooling System: MODES 1,2, and 3.

ACTION:

1. Modes 1, 2, and 3 with Pressurizer Pressure ≥ 1750 psia:

a. With one containment spray train inoperable, restore the inoperable spray train to OPERABLE status within 72 hours and within 10 days from initial discovery of failure to meet the LCO; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 54 hours.

b. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 7 days and within 10 days from initial discovery of failure to meet the LCO; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

c. With one containment spray train and one containment cooling train inoperable, concurrently implement ACTIONS a. and b. The completion intervals for ACTION a. and ACTION b. shall be tracked separately for each train starting from the time each train was discovered inoperable.

d. With two containment cooling trains inoperable, restore one cooling train to OPERABLE status within 72 hours; otherwise be in MODE 3 within the next 6 hours and in MODE 4 within the following 6 hours.

e. With two containment spray trains inoperable or any combination of three or more trains inoperable, enter LCO 3.0.3 immediately.

2. Mode 3 with Pressurizer Pressure < 1750 psia:

a. With one containment cooling train inoperable, restore the inoperable cooling train to OPERABLE status within 72 hours; otherwise be in MODE 4 within the next 6 hours.

b. With two containment cooling trains inoperable, enter LCO 3.0.3 immediately

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

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

b. By verifying, that on recirculation flow, each pump develops a discharge pressure of greater than or equal to 200 psig when tested pursuant to the Inservice Testing Program.

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 CSAS test signal.

2. Verifying that upon a Recirculation Actuation Test Signal (RAS), the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each spray pump starts automatically on a CSAS test signal.

d. At least once per 10 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

4.6.2.1.1 Each containment cooling train shall be demonstrated OPERABLE:

a. At least once per 31 days by:

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

2. Verifying a cooling water flow rate of greater than or equal to 1200 gpm to each cooling unit.

b. At least once per 18 months, during shutdown, by verifying that each containment cooling train starts automatically on an SIAS test signal.

IODINE REMOVAL SYSTEM (IRS)

LIMITING CONDITION FOR OPERATION

3.6.2.2 The Iodine Removal System shall be OPERABLE with:

a. A hydrazine storage tank containing a minimum volume of 675 gallons of ≥ 25.4% by weight N_2H_4 (Hydrazine) solution, and

b. Two iodine removal pumps each capable of adding N_2H_4 solution from the hydrazine storage tank to a containment spray system pump flow.

APPLICABILITY: MODES 1, 2 and 3*.

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 Iodine Removal 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 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. The above required iodine removal pumps shall be demonstrated OPERABLE by verifying a flow rate of between 0.71 gpm and 0.82 gpm when tested pursuant to the Inservice Testing Program.

c. At least once per 6 months by:

1. Verifying the contained solution volume in the tank, and

2. Verifying the concentration of the N_2H_4 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 CSAS test signal.

* Applicable only when pressurizer pressure is ≥ 1750 psia.

DELETED

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more containment isolation valve(s) inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and either:

a. Restore the inoperable valve(s) to OPERABLE status within 4 hours, or

b. Isolate each affected penetration within 4 hours by use of at least one deactivated automatic valve secured in the isolation position, or

c. Isolate each affected penetration within 4 hours by use of at least one closed manual valve or blind flange; or

d. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The containment isolation valves shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.2 Each automatic containment isolation valve shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a Containment Isolation test signal (CIAS) and/or a Safety Injection test signal (SIAS), each isolation valve actuates to its isolation position.

b. Verifying that on a Containment Radiation-High test signal, each containment purge valve actuates to its isolation position.

4.6.3.3 The isolation time of each power-operated or automatic containment isolation valve shall be determined to be within its limit when tested pursuant to the Inservice Testing Program.

Pages 3/4 6-22 through 3/4 6-23 have been DELETED.

Page 3/4 6-24 is the next valid page.

CONTAINMENT SYSTEMS

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

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 demonstrate within the next 24 hours that the grab sample system of the inoperable hydrogen analyzer has the capability to draw a sample of the containment atmosphere into the grab sample canister. Verify this capability of the grab sample system at least once per 30 days thereafter. Return the inoperable hydrogen analyzer to OPERABLE status within an additional 60 days. Otherwise, be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analyzer shall be demonstrated OPERABLE by the performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days, and at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. Nominally one volume percent hydrogen, balance nitrogen and oxygen.

b. Nominally four volume percent hydrogen, balance nitrogen and oxygen.

CONTAINMENT SYSTEMS

ELECTRIC HYDROGEN RECOMBINERS - W

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.).

   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.

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5 Two vacuum relief lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one vacuum relief line inoperable, restore the vacuum relief line 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

4.6.5 Verify each vacuum relief line OPERABLE in accordance with the Inservice Testing Program.

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM (SBVS)

LIMITING CONDITION FOR OPERATION

3.6.6.1 Two independent Shield Building Ventilation Systems shall be OPERABLE.

APPLICABILITY: At all times in MODES 1, 2, 3, and 4.
In addition, during movement of recently irradiated fuel assemblies or during crane operations with loads over recently irradiated fuel assemblies in the Spent Fuel Storage Pool in MODES 5 and 6.

ACTION:

a. With the SBVS inoperable solely due to loss of the SBVS capability to provide design basis filtered air evacuation from the Spent Fuel Pool area, only ACTION-c is required. If the SBVS is inoperable for any other reason, concurrently implement ACTION-b and ACTION-c.

b. (1) With one SBVS inoperable in MODE 1, 2, 3, or 4, restore the inoperable system to OPERABLE status within 7 days; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

(2) With both SBVS inoperable in MODE 1, 2, 3, or 4, immediately enter LCO 3.0.3.

c. (1) With one SBVS inoperable in any MODE, restore the inoperable system to OPERABLE status within 7 days; otherwise, suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

(2) With both SBVS inoperable in any MODE, immediately suspend movement of recently irradiated fuel assemblies within the Spent Fuel Storage Pool and crane operations with loads over recently irradiated fuel in the Spent Fuel Storage Pool.

SURVEILLANCE REQUIREMENTS

4.6.6.1 Each Shield Building Ventilation System shall be demonstrated OPERABLE:

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

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

1. Performing a visual examination of SBVS in accordance with ANSI N-510-1980.

SURVEILLANCE REQUIREMENTS (continued)

2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ANSI N-510-1980. The distribution shall be ± 20% of the average flow per unit.

3. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

4. Verifying that the HEPA filter banks remove ≥ 99.825% of the DOP when they are tested in place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

5. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N-510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a 2-inch laboratory sample from the installed sample canisters demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

d. At least once per 18 months by:

1. Verifying that the pressure drop across the demisters, electric heaters, HEPA filters, and charcoal adsorber banks is less than 8.5 inches Water Gauge (WG) while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.

3. Verifying that the filter cooling makeup and cross connection valves can be manually opened.

4. Verifying that each system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

5. Verifying that the main heaters dissipate 30 ± 3 kW and the auxiliary heaters dissipate 1.5 ± 0.25 kW when tested in accordance with ANSI N-510-1980.

SURVEILLANCE REQUIREMENTS (Continued)

6. Verifying that each system achieves a negative pressure of greater than 0.125 inch WG in the fuel storage building after actuation of a fuel storage building high radiation test signal.

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 99.825% of the DOP when they are tested in-place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 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% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

CONTAINMENT SYSTEMS

SHIELD BUILDING INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.2 SHIELD BUILDING INTEGRITY shall be maintained.

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

ACTION:

Without SHIELD BUILDING INTEGRITY, restore SHIELD BUILDING INTEGRITY 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

4.6.6.2 SHIELD BUILDING INTEGRITY shall be demonstrated at least once per 31 days by verifying that the door in each access opening is closed except when the access opening is being used for normal transit entry and exit.

CONTAINMENT SYSTEMS

SHIELD BUILDING STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.3 The structural integrity of the shield building shall be maintained at a level consistent with the acceptance criteria in Surveillance Requirement 4.6.6.3.

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

ACTION:

With the structural integrity of the shield building 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

4.6.6.3 The structural integrity of the shield building shall be determined, in accordance with the Containment Leakage Rate Testing Program, by a visual inspection of the exposed accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation.

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as shown in Table 3.7-2.

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 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 Verify each main steam line code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within +/- 1% of 1000 psia for valves 8201 through 8208, and within +/- 1% of 1040 psia for valves 8209 through 8216 specified in Table 3.7-2.

TABLE 3.7-1

MAXIMUM ALLOWABLE 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 Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	92.8
2	79.6
3	66.3

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER		LIFT SETTING (+ 1% to - 3%)
	Header A	Header B	
a.	8201	8205	$\geq$ 955.3 psig and $\leq$ 995.3 psig
b.	8202	8206	$\geq$ 955.3 psig and $\leq$ 995.3 psig
c.	8203	8207	$\geq$ 955.3 psig and $\leq$ 995.3 psig
d.	8204	8208	$\geq$ 955.3 psig and $\leq$ 995.3 psig
e.	8209	8213	$\geq$ 994.1 psig and $\leq$ 1035.7 psig
f.	8210	8214	$\geq$ 994.1 psig and $\leq$ 1035.7 psig
g.	8211	8215	$\geq$ 994.1 psig and $\leq$ 1035.7 psig
h.	8212	8216	$\geq$ 994.1 psig and $\leq$ 1035.7 psig

LIMITING CONDITION FOR OPERATION

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

a. Two feedwater pumps, each capable of being powered from separate OPERABLE emergency busses, and

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

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each motor-driven pump develops a discharge pressure of greater than or equal to 1270 psig on recirculation flow.

2. Verifying that the turbine-driven pump develops a discharge pressure of greater than or equal to 1260 psig on recirculation flow when the secondary steam supply pressure is greater than 50 psig. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

3. 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.

SURVEILLANCE REQUIREMENTS (Continued)

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 auxiliary feedwater actuation test signal.

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

c. Following an extended cold shutdown (30 days or longer) and prior to entering MODE 2, a flow test shall be performed to verify the normal flow path from the condensate storage tank (CST) to the steam generators.

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST #2) shall be OPERABLE with a contained volume of at least 307,000 gallons.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With the condensate storage tank inoperable, within 4 hours restore the CST to OPERABLE status or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the contained water volume is within its limits when the tank is the supply source for the auxiliary feedwater pumps.

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be less than or equal to 0.10 microcuries/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system greater than 0.10 microcuries/gram DOSE EQUIVALENT I-131, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.4 The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.7-1.

TABLE 4.7-1

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Activity Determination	3 Times Per 7 Days With A Maximum Time of 72 Hours Between Samples
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

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable but open, POWER OPERATION may continue provided the inoperable valve is restored to OPERABLE status within 4 hours; otherwise, be in at least HOT STANDBY within the next 6 hours.

MODES 2, 3- and 4 With one or both main steam line isolation valve(s) inoperable, subsequent operation in MODES 2, 3 or 4 may proceed provided the isolation valve(s) is (are) maintained closed. Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 24 hours.

The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 6.75 seconds when tested pursuant to the Inservice Testing Program.

MAIN FEEDWATER ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Four main feedwater isolation valves (MFIVs) shall be OPERABLE.

APPLICABILITY:* MODES 1, 2 and 3, except when the MFIV is closed and deactivated.

ACTION:

a. With one MFIV inoperable in one or more main feedwater lines, OPERATION may continue provided each inoperable valve is restored to OPERABLE status, closed, or isolated within 72 hours. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With two MFIVs inoperable in the same flowpath, restore at least one of the inoperable MFIVs to OPERABLE status or close one of the inoperable valves within 4 hours. Otherwise, be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6.a Each MFIV shall be demonstrated OPERABLE by verifying full closure within 5.15 seconds when tested pursuant to the Inservice Testing Program. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

4.7.1.6.b For each inoperable MFIV, verify that it is closed or isolated once per 7 days.

* Each MFIV shall be treated independently.

ATMOSPHERIC DUMP VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.7 The atmospheric dump and associated block valves shall be OPERABLE with:

a. All atmospheric dump valves in manual control above 15% of RATED THERMAL POWER, and

b. No more than one atmospheric dump valve per steam generator in automatic control below 15% of RATED THERMAL POWER.

APPLICABILITY: MODE 1.

ACTION:

a. With less than one atmospheric dump and associated block valve per steam generator OPERABLE, restore the required atmospheric dump and associated block valve to OPERABLE status within 72 hours, or be in at least HOT STANDBY within the next 6 hours.

b. With more than the permissible number of atmospheric dump valves in automatic control, return the atmospheric dump valves to manual control within 1 hour, or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 Each atmospheric dump valve shall be verified to be in the manual operation mode at least once per 24 hours during operation at $\geq$ 15% of RATED THERMAL POWER.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2 The temperature of the secondary coolant in the steam generators shall be greater than 100°F when the pressure of the secondary 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 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 of the secondary side of the steam generators shall be determined to be less than 200 psig at least once per hour when the temperature of the secondary coolant is less than 100°F.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3 At least two independent component cooling water loops shall be OPERABLE.*

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

ACTION:

With only one component cooling water loop OPERABLE, restore at least two loops to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 At least two component cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated or automatic) servicing safety-related equipment that is not locked, sealed, or otherwise secured in position, is in its correct position.

b. At least once per 18 months during shutdown, by verifying that each automatic valve servicing safety-related equipment actuates to its correct position on an SIAS test signal.

*When CCW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves shall be verified to be consistent with the appropriate power supply at least once per 24 hours. Upon receipt of annunciation for improper alignment of the pump 2C motor power in relation to any of its motor-operated discharge valves positions, restore proper system alignment within 2 hours.

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With only one intake 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

4.7.4 At least two intake 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.

*When ICW pump 2C is being used to satisfy the requirements of this specification, the alignment of the discharge valves must be verified to be consistent with the appropriate power supply at least once per 24 hours.

3/4.7.5 ULTIMATE HEAT SINK

LIMITING CONDITION FOR OPERATION

3.7.5.1 The ultimate heat sink shall be OPERABLE with:

a. Cooling water from the Atlantic Ocean providing a water level above -10.5 feet elevation, Mean Low Water, at the plant intake structure, and

b. Two OPERABLE valves in the barrier dam between Big Mud Creek and the intake structure.

APPLICABILITY: At all times.

ACTION:

a. With the water level requirement of the above specification not satisfied, be in at least HOT STANDBY within 6 hours and provide cooling water from Big Mud Creek within the next 12 hours.

b. With one isolation valve in the barrier dam between Big Mud Creek and the intake structure inoperable, restore the inoperable valve to OPERABLE status within 72 hours, or within the next 24 hours, install a temporary flow barrier and open the barrier dam isolation valve. The availability of the onsite equipment capable of removing the barrier shall be verified at least once per 7 days thereafter.

c. With both of the isolation valves in the barrier dam between the intake structure and Big Mud Creek inoperable, within 24 hours, either:

   1. Install both temporary flow barriers and manually open both barrier dam isolation valves. The availability of the onsite equipment capable of removing the barriers shall be verified at least once per 7 days thereafter, or

   2. Be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

4.7.5.1.2 The isolation valves in the barrier dam between the intake structure and Big Mud Creek shall be demonstrated OPERABLE at least once per 6 months by cycling each valve through at least one complete cycle of full travel.

PLANT SYSTEMS

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for the facility site via stoplogs which shall be installed on the southside of the RAB and the southernmost door on east wall whenever a hurricane warning for the plant is posted.

APPLICABILITY: At all times.

ACTION:

With a Hurricane Watch issued for the facility site, ensure the stoplogs are removed from storage and are prepared for installation. The stoplogs shall be installed anytime a Hurricane Warning is posted.

SURVEILLANCE REQUIREMENTS

4.7.6.1 Meteorological forecasts shall be obtained from the National Hurricane Center in Miami, Florida at least once per 6 hours during either a Hurricane Watch or a Hurricane Warning.

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM (CREACS)

LIMITING CONDITION FOR OPERATION

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

a. A filter train and its associated fan per system, and

b. At least one air conditioning unit per system, and

c. Two isolation valves in the kitchen area exhaust duct, and

d. Two isolation valves in the toilet area exhaust duct, and

e. Two isolation valves in each (North and South) air intake duct.

APPLICABILITY: ALL MODES.

ACTION:

MODES 1, 2, 3, and 4:

a. 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 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With both control room emergency air cleanup systems inoperable, restore at least one system to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the next 30 hours.

c. With an isolation valve in an air intake duct or air exhaust duct inoperable, operation may continue provided the other isolation valve in the same air intake or air exhaust duct is maintained closed; otherwise be in at least HOT STANDBY in the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

MODES 5 and 6:

a. With one control room emergency air cleanup system inoperable, restore the inoperable system to OPERABLE status within 7 days or initiate and maintain operation of the remaining OPERABLE control room emergency air cleanup system in the recirculation mode.

b. With both control room emergency air cleanup systems inoperable, suspend all operations involving CORE ALTERATIONS or positive reactivity changes*.

c. With an isolation valve in an air intake duct or air exhaust duct inoperable, maintain the other isolation valve in the same air intake or air exhaust duct closed or suspend any core alterations or positive reactivity addition operations.

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* Limited plant cooldown or boron dilution is allowed provided the change is accounted for in the calculated SHUTDOWN MARGIN.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

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

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.

b. At least once per 31 days by (1) initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes and (2) starting, unless already operating each air conditioning unit and verifying that it operates for at least 8 hours.

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

   1. Performing a visual examination of CREACS in accordance with ANSI N-510-1980.

   2. Performing air flow distribution to HEPA filters and charcoal adsorbers in accordance with ANSI N-510-1980. The distribution shall be $\pm$ 20% of the average flow per unit.

   3. Verifying that the charcoal adsorbers remove $\geq$ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in accordance with ANSI N-510-1980 while operating the system at 2000 cfm $\pm$ 10%.

   4. Verifying that the HEPA filters remove $\geq$ 99.95% of the DOP when they are tested in accordance with ANSI N-510-1980 while operating the system at 2000 cfm $\pm$ 10%.

   5. Verifying a system flow rate of 2000 cfm $\pm$10%.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canisters demonstrates a removal efficiency of $\geq$ 99.825% for methyl iodide when tested in accordance with ASTM D3803-1989 (30°C, 95% RH).

e. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined prefilters, HEPA filters and charcoal adsorber banks is less than 7.4 inches Water Gauge while operating the system at a flow rate of 2000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

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

3. Verifying that the system maintains the control room at a positive pressure of greater than or equal to 1/8 inch Water Gauge relative to the outside atmosphere during system operation with $\leq$ 450 cfm outside air intake.

4. Verifying that on a containment isolation test signal from Unit 1 the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove greater than or equal to 99.95% of the DOP when they are tested in-place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 2000 cfm $\pm$ 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove greater than or equal to 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 2000 cfm $\pm$ 10%.

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8 Two independent ECCS area ventilation systems shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.8 Each ECCS area ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating from the control room and verifying that the system operates for at least 15 minutes.

b. At least once per 18 months by:

   1. Verifying a system flow rate of 30,000 cfm ± 10% during system operation.

   2. Verifying that the system starts on a safety injection actuation test signal.

3/4 7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All safety-related snubbers 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 safety related snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9f. on the supported component or declare the supported system inoperable and follow the appropriate ACTION statement for that system.

SURVEILLANCE REQUIREMENTS

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

a. Inspection Types

As used in this specification, "type of snubber" shall mean snubbers of the same design and manufacturer, irrespective of capacity.

b. Visual Inspections

Snubbers are categorized as inaccessible or accessible during reactor operation. Each of these categories (inaccessible or accessible) may be inspected independently according to the schedule determined by Table 4.7-2. The visual inspection interval for each category of snubber shall be determined based upon the criteria provided in Table 4.7-2 and the first inspection interval determined using this criteria shall be based upon the previous inspection interval as established by the requirements in effect before Amendment

SURVEILLANCE REQUIREMENTS (Continued)

c. Visual Inspection Acceptance Criteria

Visual inspections shall verify that (1) the snubber has no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are functional, and (3) fasteners for the attachment of the snubber to the component and to the snubber anchorage are functional. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that (i) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and (ii) the affected snubber is functionally tested in the as-found condition and determined OPERABLE per specification 4.7.10.e. All snubbers found connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

TABLE 4.7-2
SNUBBER VISUAL INSPECTION INTERVAL

	NUMBER OF UNACCEPTABLE SNUBBERS		
Population or Category (Notes 1 and 2)	Column A Extend Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

SURVEILLANCE REQUIREMENTS (Continued)

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category sizes and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as determined by interpolation.

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number of unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one-third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Columns B and C.

Note 6: The provisions of Specification 4.0.2 are applicable for all inspection intervals up to and including 48 months.

d. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of either: (1) At least 10% of the total of each type of safety related snubber in use in the plant shall be functionally tested either in place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.9e. an additional 10% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested or (2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.9e. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7-1. If at any time the point plotted falls in the "Reject" region, all snubbers of that type design 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 of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of snubbers of each type. The representative sample should be weighted to include more snubbers from severe service areas such as near heavy equipment. Snubbers placed in the same location as snubbers which failed the previous functional test shall be included in the next test lot if the failure analysis shows that failure was due to location.

e. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression.

2. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range.

3. Where required, the force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel.

SURVEILLANCE REQUIREMENTS (Continued)

Functional Test Acceptance Criteria (Continued)

4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.

5. Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

f. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and if caused by manufacturer or design deficiency all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Specification 4.7.9.e for snubbers not meeting the functional test acceptance criteria.

g. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

h. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals do not fail between surveillance inspections. The maximum expected service life for the various seals, seal materials, and applications shall be estimated based on engineering information and the seals shall be replaced so that the maximum expected service life does not expire during a period when the snubber is required to be OPERABLE. The seal replacements shall be documented and the documentation shall be retained in accordance with Specification 6.10.2.

i. Exemption From Visual Inspection or Functional Tests

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the Commission if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date.

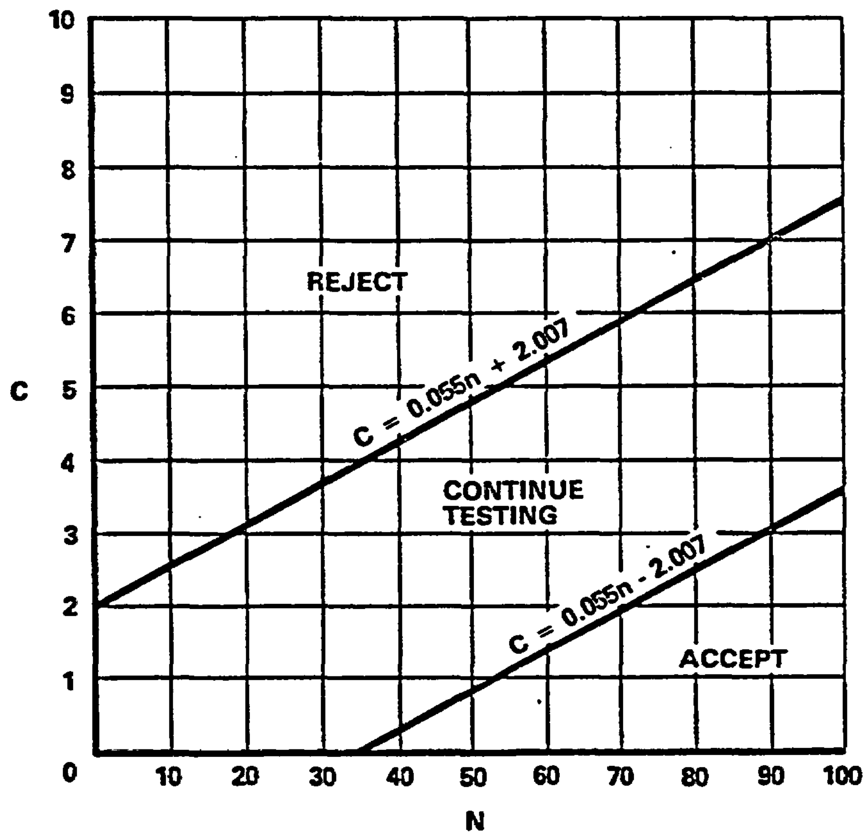

FIGURE 4.7-1 SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

TABLE 3.7-3a

DELETED

TABLE 3.7-3b

DELETED

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 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

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

a. The licensee, or

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

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

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

a. Sources in use - At least once per 6 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.

SURVEILLANCE REQUIREMENTS (Continued)

b. Stored sources not in use - Each sealed source and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous 6 months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

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

4.7.10.3 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.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators, each with:

1. Two separate engine-mounted fuel tanks containing a minimum volume of 200 gallons of fuel each,

2. A separate fuel storage system containing a minimum volume of 40,000 gallons of fuel, and

3. A separate fuel transfer pump.

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

ACTION:

a. With one offsite circuit of 3.8.1.1.a inoperable, except as provided in Action f. below, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. Restore the offsite circuit to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

b. With one diesel generator of 3.8.1.1.b inoperable, demonstrate the OPERABILITY of the A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG*; restore the diesel generator to OPERABLE status within 14 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

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* If the absence of any common-cause failure cannot be confirmed, this test shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ACTION: (Continued)

c. With one offsite A.C. circuit and one diesel generator inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within one hour and at least once per 8 hours thereafter; and if the EDG became inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned preventative maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE EDG by performing Surveillance Requirement 4.8.1.1.2a.4 within 8 hours, unless it can be confirmed that the cause of the inoperable EDG does not exist on the remaining EDG*. 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 the other A.C. power source (offsite circuit or diesel generator) to OPERABLE status in accordance with the provisions of Section 3.8.1.1 ACTION Statement a or b, as appropriate, with the time requirement of that ACTION Statement based on the time of the initial loss of the remaining inoperable A.C. power source. Additionally, within 4 hours from the discovery of concurrent inoperability of required redundant feature(s) (including the steam driven auxiliary feed pump in MODE 1, 2, and 3), declare required feature(s) supported by the inoperable EDG inoperable if its redundant required feature(s) is inoperable.

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* If the absence of any common-cause failure cannot be confirmed, this test shall be completed regardless of when the inoperable EDG is restored to OPERABILITY.

ACTION: (Continued)

d. With two of the required offsite A.C. circuits inoperable, restore one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. Following restoration of one offsite source, follow ACTION Statement a. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable offsite A.C. circuit.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter; restore 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. Following restoration of one diesel generator unit, follow ACTION Statement b. with the time requirement of that ACTION Statement based on the time of initial loss of the remaining inoperable diesel generator.

f. With one Unit 2 startup transformer (2A or 2B) inoperable and with a Unit 1 startup transformer (1A or 1B) connected to the same A or B offsite power circuit and administratively available to both units, then should Unit 1 require the use of the startup transformer administratively available to both units, Unit 2 shall demonstrate the operability of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter. Restore the inoperable startup transformer to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each of the above required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying correct breaker alignments, indicated power availability; and

b. Demonstrated OPERABLE at least once per 18 months 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. At least once per 31 days on a STAGGERED TEST BASIS BY:

1. Verifying the fuel level in the engine-mounted 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 engine-mounted tank,
4. Verifying the diesel starts from ambient condition and accelerates to approximately 900 rpm in less than or equal to 10 seconds **. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal**. The diesel generator shall be started for this test by using one of the following signals:
   a) Manual/Local.
   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 3685 kW in accordance with the manufacturer's recommendations, and operates within a load band of 3450 to 3685 kW*** at least an additional 60 minutes, and
6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. By removing accumulated water:
   1. From the engine-mounted fuel tank at least once per 31 days and after each occasion when the diesel is operated for greater than 1 hour, and
   2. From the storage tank at least once per 31 days.

** The diesel generator start (10 sec) from ambient conditions shall be performed at least once per 184 days in these surveillance tests. All other diesel generator starts for purposes of this surveillance testing may be preceded by an engine prelube period and may also include warmup procedures (e.g., gradual acceleration) as recommended by the manufacturer so that mechanical stress and wear on the diesel generator is minimized.

*** The indicated load band is meant as guidance to avoid routine overloading. Variations in loads in excess of the band due to changing bus loads shall not invalidate this test.

SURVEILLANCE REQUIREMENTS (continued)

c. By sampling new fuel all in accordance with ASTM D4057-81 prior to addition to the storage tanks and:

1. By verifying in accordance with the tests specified in ASTM D975-81 prior to addition to the storage tanks that the sample has:

a) An API Gravity of within 0.3 degrees at 60°F or a specific gravity of within 0.0016 at 60/60°F, when compared to the supplier's certificate or an absolute specific gravity at 60/60°F of greater than or equal to 0.83 but less than or equal to 0.89 or an API gravity of 60°F of greater than or equal to 27 degrees but less than or equal to 39 degrees.

b) A kinematic viscosity at 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes, if gravity was not determined by comparison with the supplier's certification.

c) A flash point equal to or greater than 125°F, and

d) A clear and bright appearance with proper color when tested in accordance with ASTM D4176-82.

2. By verifying within 31 days of obtaining the sample that the other properties specified in Table 1 of ASTM D975-81 are met when tested in accordance with ASTM D975-81 except that the analysis for sulfur may be performed in accordance with ASTM D1552-79 or ASTM D2622-82.

d. At least once every 31 days by obtaining a sample of fuel oil from the storage tanks in accordance with ASTM D2276-83 and verifying that total particulate contamination is less than 10 mg/liter when checked in accordance with ASTM D2276-83, Method A, or Annex A-2.

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

1. DELETED

2. Verifying generator capability to reject a load of greater than or equal to 453 kW while maintaining voltage at 4160 ± 420 volts and frequency at 60 ± 1.2 Hz.

3. Verifying the generator capability to reject a load of 3685 kW without tripping. The generator voltage shall not exceed 4784 volts during and following the load rejection.

4. Simulating a loss-of-offsite power by itself, and:

   a) Verifying deenergization of the emergency busses and load shedding from the emergency busses.

   b) Verifying the diesel starts on the auto-start signal, **** energizes the emergency busses with permanently connected loads within 10 seconds, energizes the auto-connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the shutdown loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

5. Verifying that on an ESF actuation test signal (without loss-of-offsite power) the diesel generator starts**** on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The steady-state generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the auto-start signal; the generator voltage and frequency shall be maintained within these limits during this test.

6. Simulating a loss-of-offsite power in conjunction with an ESF actuation test signal, and

   a) Verifying deenergization 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 emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ± 420 volts and 60 ± 1.2 Hz during this test.

****This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

SURVEILLANCE REQUIREMENTS (Continued)

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

7. Verifying the diesel generator operates for at least 24 hours.[****] During the first 2 hours of this test, the diesel generator shall be loaded within a load band of 3800 to 3985 kW[#] and during the remaining 22 hours of this test, the diesel generator shall be loaded within a load band of 3450 to 3685 kW[#]. The generator voltage and frequency shall be 4160 ± 420 volts and 60 ± 1.2 Hz within 10 seconds after the start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test.

8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3935 kW.

9. 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 load to the offsite power source, and

   c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted tanks of each diesel via the installed cross connection lines.

\# This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

SURVEILLANCE REQUIREMENTS (Continued)

12. Verifying that the automatic load sequence timers are operable with the interval between each load block within ±1 second of its design interval.

13. Performing Surveillance Requirement 4.8.1.1.2a.4 within 5 minutes of shutting down the diesel generator after it has operated within a load band of 3450 kW to 3685 kW# for at least 2 hours or until operating temperatures have stabilized.

f. 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 approximately 900 rpm in less than or equal to 10 seconds.

g. At least once per 10 years by:

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

2. Performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with the Inservice Inspection Program.

4.8.1.1.3 Reports – (Not Used).

4.8.1.1.4 The Class 1E underground cable system shall be demonstrated OPERABLE within 30 days after the movement of any loads in excess of 80% of the ground surface design basis load over the cable ducts by pulling a mandrel with a diameter of at least 80% of the duct's inside diameter through a duct exposed to the maximum loading (duct nearest the ground's surface) and verifying that the duct has not been damaged.

This band is meant as guidance to avoid routine overloading of the engine. Variations in load in excess of this band due to changing bus loads shall not invalidate this test.

**** This test may be conducted in accordance with the manufacturer's recommendations concerning engine prelube period.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

(NOT USED)

ELECTRICAL POWER SYSTEMS

A.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Two engine-mounted fuel tanks containing a minimum volume of 200 gallons of fuel,

2. A fuel storage system containing a minimum volume of 40,000 gallons of fuel, and

3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, movement of irradiated fuel, or crane operation with loads over the fuel storage pool, and within 8 hours, depressurize and vent the Reactor Coolant System through a greater than or equal to 3.58 square inch vent. In addition, when in MODE 5 with the reactor coolant loops not filled, or in MODE 6 with the water level less than 23 feet above the reactor vessel flange, immediately initiate corrective action to restore the required sources to OPERABLE status as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.1.2.1 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 (except for requirement 4.8.1.1.2a.5).

ELECTRICAL POWER SYSTEMS

3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt Battery bank No. 2A and a full capacity charger.

b. 125-volt Battery bank No. 2B and a full capacity charger.

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

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1 within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

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

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110 volts, or battery overcharge with battery terminal voltage above 150 volts, by verifying that:

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

   2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

   3. The average electrolyte temperature of 10% (60 cells total) of connected cells is above 50°F.

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

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

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

   3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

   4. The battery charger will supply at least 300 amperes at 140 volts for at least 6 hours.

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

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. This performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

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 REQUIREMENT

	CATEGORY A[1]	CATEGORY B[2]	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable[3] value for each connected cell
Electrolyte Level	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	>Minimum level indication mark, and $\leq$ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	$\geq$ 2.13 volts	$\geq$ 2.13 volts[c]	> 2.07 volts
Specific Gravity[a]	$\geq$ 1.195[b]	$\geq$ 1.190 Average of all connected cells > 1.200	Not more than .020 below the average of all connected cells Average of all connected cells $\geq$ 1.190[b]

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

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, one 125-volt battery bank and a full capacity charger shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery bank inoperable, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel; initiate corrective action to restore the required battery bank to OPERABLE status as soon as possible, and within 8 hours, depressurize and vent the Reactor Coolant System through a 3.58 square inch vent.

b. With the required full capacity charger inoperable, demonstrate the OPERABILITY of its associated battery banks by performing Surveillance Requirement 4.8.2.1a.1. within 1 hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery bank and charger shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner with both tie breakers open between redundant busses and between St. Lucie Unit 1 and Unit 2.

a. Train A A.C. Emergency Busses consisting of:

1. 4160 volt Emergency Bus # 2A3
2. 480 volt Emergency Bus # 2A2
3. 480 volt Emergency Bus # 2A5
4. 480 volt MCC Emergency Bus # 2A5
5. 480 volt MCC Emergency Bus # 2A6
6. 480 volt MCC Emergency Bus # 2A7
7. 480 volt MCC Emergency Bus # 2A8
8. 480 volt MCC Emergency Bus # 2A9

b. Train B A.C. Emergency Busses consisting of:

1. 4160 volt Emergency Bus # 2B3
2. 480 volt Emergency Bus # 2B2
3. 480 volt Emergency Bus # 2B5
4. 480 volt MCC Emergency Bus #2B5
5. 480 volt MCC Emergency Bus #2B6
6. 480 volt MCC Emergency Bus #2B7
7. 480 volt MCC Emergency Bus #2B8
8. 480 volt MCC Emergency Bus #2B9

c. 120 volt A.C. Instrument Bus # 2MA energized from its associated inverter connected to D.C. Bus # 2A*.

d. 120 volt A.C. Instrument Bus # 2MB energized from its associated inverter connected to D.C. Bus # 2B*.

e. 120 volt A.C. Instrument Bus # 2MC energized from its associated inverter connected to D.C. Bus # 2A*.

f. 120 volt A.C. Instrument Bus # 2MD energized from its associated inverter connected to D.C. Bus # 2B*.

g. 125 volt D.C. Bus #2A energized from Battery Bank #2A.

h. 125 volt D.C. Bus #2B energized from Battery Bank #2B.

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

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*Two inverters may be disconnected from their D.C. Bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on their associated battery bank provided (1) their vital busses are energized, and (2) the vital busses associated with the other battery bank are energized from their associated inverters and connected to their associated D.C. Bus.

ACTION:

a. With one of the required trains of A.C. Emergency busses not fully energized, re-energize the train within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Instrument Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Instrument Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours and (2) re-energize the A.C. Instrument Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. Bus not energized from its associated Battery Bank, re-energize the D.C. Bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized and in the specified manner:

a. One train of A.C. emergency busses consisting of one 4160 volt and two 480 volt A.C. emergency busses.

b. Two 120 volt A.C. Instrument Busses energized from their associated inverters connected to their respective D.C. busses.

c. One 125 volt D.C. bus energized from its associated battery bank.

APPLICABILITY: MODES 5 and 6.

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, operations involving positive reactivity additions that could result in loss of required SHUTDOWN MARGIN or boron concentration, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible, and within 8 hours depressurize and vent the RCS through a 3.58 square inch vent.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES

LIMITING CONDITION FOR OPERATION

3.8.4 The thermal overload protection bypass devices, integral with the motor starter, of each valve listed in Table 3.8-1 shall be OPERABLE.

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

4.8.4 The above required thermal overload protection bypass devices shall be demonstrated OPERABLE.

a. At least once per 18 months, by visually verifying the bypass switch to be in the bypass position for those thermal overload devices which are either:

   1. Continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, or

   2. Normally in force during plant operation and bypassed under accident conditions.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION of a representative sample of at least 25% of:

   1. All thermal overload devices which are not bypassed, such that each non-bypassed device is calibrated at least once per 6 years.

   2. All thermal overload devices which are continuously bypassed and temporarily placed in force only when the valve motors are undergoing periodic or maintenance testing, and thermal overload devices normally in force and bypassed under accident conditions such that each thermal overload is calibrated and each valve is cycled through at least one complete cycle of full travel with the motor-operator when the thermal overload is OPERABLE and not bypassed, at least once per 6 years.

TABLE 3.8-1

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES

VALVE NUMBER	FUNCTION	BYPASS (YES/NO)
RCS: V-1476	LTOP ISOLATION	YES
V-1477	LTOP ISOLATION	YES
CVCS:V-2508	BAMT ISOL.	YES
V-2509	BAMT ISOL.	YES
V-2514	BAMP DISCH.	YES
V-2525	PMW SUPPLY	YES
V-2553	CHARGING PUMP BYPASS	YES
V-2554	CHARGING PUMP BYPASS	YES
V-2555	CHARGING PUMP BYPASS	YES
V-2501	VCT ISOL.	YES
V-2504	RWT ISOL.	YES
SIS: FCV-3301	SHUTDOWN COOLING	YES
FCV-3306	SHUTDOWN COOLING	YES
HCV-3512	SHUTDOWN COOLING	YES
HCV-3657	SHUTDOWN COOLING	YES
V-3456	SDC ISOL.	YES
V-3457	SDC ISOL.	YES
V-3517	SDC ISOL.	YES
V-3658	SDC ISOL.	YES
V-3540	HOT LEG INJECTION	YES
V-3550	HOT LEG INJECTION	YES
V-3523	HOT LEG INJECTION	YES
V-3551	HOT LEG INJECTION	YES
V-3656,3654	HPSI ISOL.	YES
V-3659	SI RECIRCULATION	YES
V-3660	SI RECIRCULATION	YES
V-3615,25,35,45	LPSI INJ.	YES
V-3616,26,36,46	HPSI INJ.	YES
V-3617,27,37,47	HPSI INJ.	YES
V-3480	SDC ISOL.	YES
V-3481	SDC ISOL.	YES
V-3651	SDC ISOL.	YES
V-3652	SDC ISOL.	YES
V-3545	SDC X-TIE	YES
V-3664	SDC ISOL.	YES
V-3665	SDC ISOL.	YES
V-3536	SDC WARMUP	YES
V-3539	SDC WARMUP	YES
V-3614,24,34,44	SIT ISOL.	YES
V-3432	RWT ISOL.	YES
V-3444	RWT ISOL.	YES

TABLE 3.8-1 (continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES

	VALVE NUMBER	FUNCTION	BYPASS (YES/NO)
MAIN STEAM:			
	MV-08-1A, 1B	MSIV BYPASS	YES
	MV-08-18A, 18B	A.D.V.	YES
	MV-08-19A, 19B	A.D.V.	YES
	MV-08-12, 13	AFW TURBINE INLET	YES
	MV-08-3	AFW TURBINE INLET	YES
	MV-08-14, 15, 16, 17	A.D.V. ISOL.	YES
MAIN FEEDWATER			
	MV-09-9, 10, 11, 12	AUX. FEED ISOL.	YES
	MV-09-13, 14	AUX. FEED X-TIE	YES
ICW:	MV-21-2, 3	ICW ISOL.	YES
CCW:	MV-14-17, 18, 19, 20	FUEL POOL ISOL.	YES
	MV-14-9, 10, 11, 12, 13, 14, 15, 16	CONT. FAN ISOL.	YES
	MV-14-1, 2, 3, 4	CCW PUMP ISOL.	YES
C.S.:	MV-07-1A, 1B	RWT ISOL.	YES
	MV-07-2A, 2B	SUMP ISOL.	YES
	MV-07-3, 4	SYSTEM ISOL.	YES
HVAC:	FCV-25-14, 15, 16, 17, 18, 19	CRECS ISOL.	YES
	FCV-25-24, 25	CRECS ISOL.	YES
	FCV-25-11, 12	SBVS ISOL.	YES
	FCV-25-35	VENT ISOL.	YES
	FCV-25-29, 34	H2 CONT. PURGE	YES
	FCV-25-30, 31	SFP EXHAUST	YES
	FCV-25-32, 33	SBVS INLET	YES

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 within the limit specified in the COLR.

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 1720 ppm boron or greater to restore boron concentration to within limits.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The boron concentration limit 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.

REFUELING OPERATIONS

3/4.9.2 INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two startup range neutron flux monitors shall be OPERABLE and 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 or not operating, immediately suspend all operations involving CORE ALTERATIONS or operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each startup range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL CHECK at least once per 12 hours,

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

3/4.9.3 DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel.

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

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.

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 isolation valve.

Note: Penetration flow path(s) providing direct access from the containment atmosphere to the outside atmosphere may be unisolated under administrative controls.

APPLICABILITY: During movement of recently irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving movement of recently irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment isolation valve within 72 hours prior to the start of and at least once per 7 days during movement of recently irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing of containment isolation valves per the applicable portions of Specification 4.6.3.2.

REFUELING OPERATIONS

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

LIMITING CONDITION FOR OPERATION

3.9.6 The manipulator crane shall be used for movement of fuel assemblies, with or without CEAs, and shall be OPERABLE with:

a. A minimum capacity of 2000 pounds, and

b. An overload cut off limit of less than or equal to 3000 pounds.

APPLICABILITY: During movement of fuel assemblies, with or without CEAs, within the reactor pressure vessel.

ACTION:

With the requirements for crane OPERABILITY not satisfied, suspend use of any inoperable manipulator crane from operations involving the movement of CEAs and fuel assemblies within the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.6 The manipulator crane used for movement of fuel assemblies, with or without CEAs, 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 2000 pounds and demonstrating an automatic load cut off before the crane load exceeds 3000 pounds.

REFUELING OPERATIONS

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 1600 pounds shall be prohibited from travel over fuel assemblies in the spent fuel storage pool.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 1600 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.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.1 At least one shutdown cooling loop shall be OPERABLE and in operation.*

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is greater than or equal to 23 feet.

ACTION:

With no shutdown cooling loop OPERABLE and in operation, suspend all operations involving an increase in reactor decay heat load or operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1 and within 1 hour initiate corrective action to return the required shutdown cooling loop to OPERABLE and operating status as soon as possible. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.1 At least once per 12 hours:

a. At least one shutdown cooling loop shall be verified to be in operation

b. The total flow rate of reactor coolant to the reactor pressure vessel shall be verified to be greater than or equal to 3000 gpm.**

---

* 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 reactor pressure vessel hot legs, provided no operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SHUTDOWN MARGIN of Technical Specification 3.9.1.

** The reactor coolant flow rate requirement may be reduced to 1850 gpm if the following conditions are satisfied before the reduced requirement is implemented: the reactor has been determined to have been subcritical for at least 125 hours, the maximum RCS temperature is $\leq$ 117°F, and the temperature of CCW to the shutdown cooling heat exchanger is $\leq$ 87°F.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.8.2 The independent shutdown cooling loops shall be OPERABLE and at least one shutdown cooling shall be in operation.

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required shutdown cooling loops OPERABLE, within 1 hour initiate corrective action to return the required loops to OPERABLE status, or to establish greater than or equal to 23 feet of water above the reactor pressure vessel flange, as soon as possible.

b. With no shutdown cooling loop in operation, suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the boron concentration of Technical Specification 3.9.1 and within 1 hour initiate corrective action to return the required shutdown cooling loop to operation. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.8.2 At least once per 12 hours:

a. At least one shutdown cooling loop shall be verified to be in operation.

b. The total flow rate of reactor coolant to the reactor pressure vessel shall be verified to be greater than or equal to 3000 gpm.*

---

* The reactor coolant flow rate requirement may be reduced to 1850 gpm if the following conditions are satisfied before the reduced requirement is implemented: the reactor has been determined to have been subcritical for at least 125 hours, the maximum RCS temperature is ≤ 117°F, and the temperature of CCW to the shutdown cooling heat exchanger is ≤ 87°F.

REFUELING OPERATIONS

3/4.9.9 CONTAINMENT ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment isolation system shall be OPERABLE.

APPLICABILITY: During movement of recently irradiated fuel within containment.

ACTION:

With the containment isolation system inoperable, close each of the containment penetrations providing direct access from the containment atmosphere to the outside atmosphere.

SURVEILLANCE REQUIREMENTS

4.9.9 The containment isolation system shall be demonstrated OPERABLE within 72 hours prior to the start of and at least once per 7 days during movement of recently irradiated fuel by verifying that containment isolation occurs on manual initiation and on a high radiation test signal from each of the containment radiation monitoring instrumentation channels.

3/4.9.10 WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During CORE ALTERATIONS.
During movement of irradiated fuel assemblies within containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend CORE ALTERATIONS and movement of irradiated fuel assemblies within containment, and immediately initiate action to restore refueling cavity water level to within limits.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment.

3/4.9.11 SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 The Spent Fuel Pool shall be maintained with:

a. The fuel storage pool water level greater than or equal to 23 ft over the top of irradiated fuel assemblies seated in the storage racks, and

b. The fuel storage pool boron concentration greater than or equal to 1720 ppm.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

a. With the water level requirement not satisfied, immediately suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

b. With the boron concentration requirement not satisfied, immediately suspend all movement of fuel assemblies in the fuel storage pool and initiate action to restore fuel storage pool boron concentration to within the required limit.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the spent fuel 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.

4.9.11.1 Verify the fuel storage pool boron concentration is within limit at least once per 7 days.

REFUELING OPERATIONS

SPENT FUEL CASK CRANE

LIMITING CONDITION FOR OPERATION

3.9.12 The maximum load which may be handled by the spent fuel cask crane shall not exceed 100 tons.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.12 The loaded weight of a spent fuel assembly cask shall be verified to not exceed 100 tons prior to attaching it to the spent fuel cask crane.

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, MTC, and SHUTDOWN MARGIN provided reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s).

APPLICABILITY: MODES 2 and 3*.

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 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 7 days prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

*Operation in MODE 3 shall be limited to 6 consecutive hours.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The moderator temperature coefficient, group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is restricted to the test power plateau which shall not exceed 85% of RATED THERMAL POWER, and

b. The limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.2.2 below.

APPLICABILITY: MODES 1 and 2.

ACTION:

With any of the limits of Specification 3.2.1 being exceeded while the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3 and 3.2.4 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

4.10.2.1 The THERMAL POWER shall be determined at least once per hour during PHYSICS TESTS in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, or 3.2.4 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 Specification 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.5, 3.1.3.6, 3.2.2, 3.2.3, or 3.2.4 are suspended.

SPECIAL TEST EXCEPTIONS

3/4.10.3 REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1 and noted requirements of Tables 2.2-1 and 3.3-1 may be suspended during the performance of startup and PHYSICS TESTS, provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints of the OPERABLE power level channels are set at less than or equal to 20% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be less than or equal to 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each wide range logarithmic and 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

3/4.10.4 CENTER CEA MISALIGNMENT

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 4.2.1.4 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.

3/4.10.5 CEA INSERTION DURING ITC, MTC, AND POWER COEFFICIENT MEASUREMENTS

LIMITING CONDITION FOR OPERATION

3.10.5 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 the limits of Specification 3.2.1 are maintained and determined as specified in Specification 4.10.5.2 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

4.10.5.1 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 3.1.3.6 are suspended and shall be verified to be within the test power plateau.

4.10.5.2 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 4.2.1.4 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and 3.1.3.6 are suspended.

Pages 3/4 11-2 through 3/4 11-13 (Amendment No. 61) have been deleted from the Technical Specifications. The next page is 3/4 11-14.

EXPLOSIVE GAS MIXTURE

LIMITING CONDITION FOR OPERATION

3.11.2.5 The concentration of oxygen in the waste gas decay tanks 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 decay tank greater than 2% by volume but less than or equal to 4% by volume, reduce the oxygen concentration to the above limits within 48 hours.

b. With the concentration of oxygen in the waste gas decay tank 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 immediately commence reduction of the concentration of oxygen to less than or equal to 2% by volume.

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

SURVEILLANCE REQUIREMENTS

4.11.2.5.1 The concentration of oxygen in the waste gas decay tank shall be determined to be within the above limits by continuously* monitoring the waste gases in the on service waste gas decay tank.

4.11.2.5.2 With the oxygen concentration in the on service waste gas decay tank greater than 2% by volume as determined by Specification 4.11.2.5.1, the concentration of hydrogen in the waste gas decay tank shall be determined to be within the above limits by gas partitioner sample at least once per 24 hours.

* When continuous monitoring capability is inoperable, waste gases shall be monitored in accordance with the actions specified for the Waste Gas Decay Tanks Explosive Gas Monitoring System in Chapter 13 of the Updated Final Safety Analysis Report.

RADIOACTIVE EFFLUENTS

GAS STORAGE TANKS

LIMITING CONDITION FOR OPERATION

3.11.2.6 The quantity of radioactivity contained in each gas storage tank shall be limited to less than or equal to 285,000 curies noble gases (considered as Xe-133).

APPLICABILITY: At all times.

ACTION:

a. With the quantity of radioactive material in any gas storage tank exceeding the above limit, immediately suspend all additions of radioactive material to the tank.

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 when reactor coolant system activity exceeds $\frac{100}{\bar{E}}$.

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 5.1-1.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 140 feet.

b. Nominal inside height = 232 feet.

c. Net free volume = 2.506 x 10^6 cubic feet.

d. Nominal thickness of vessel walls = 2 inches.

e. Nominal thickness of vessel dome = 1 inch.

f. Nominal thickness of vessel bottom = 2 inches.

5.2.1.2 SHIELD BUILDING

a. Minimum annular space = 4 feet.

b. Annulus nominal volume = 543,000 cubic feet.

c. Nominal outside height (measured from top of foundation mat to the top of the dome) = 228.5 feet.

d. Nominal inside diameter = 148 feet.

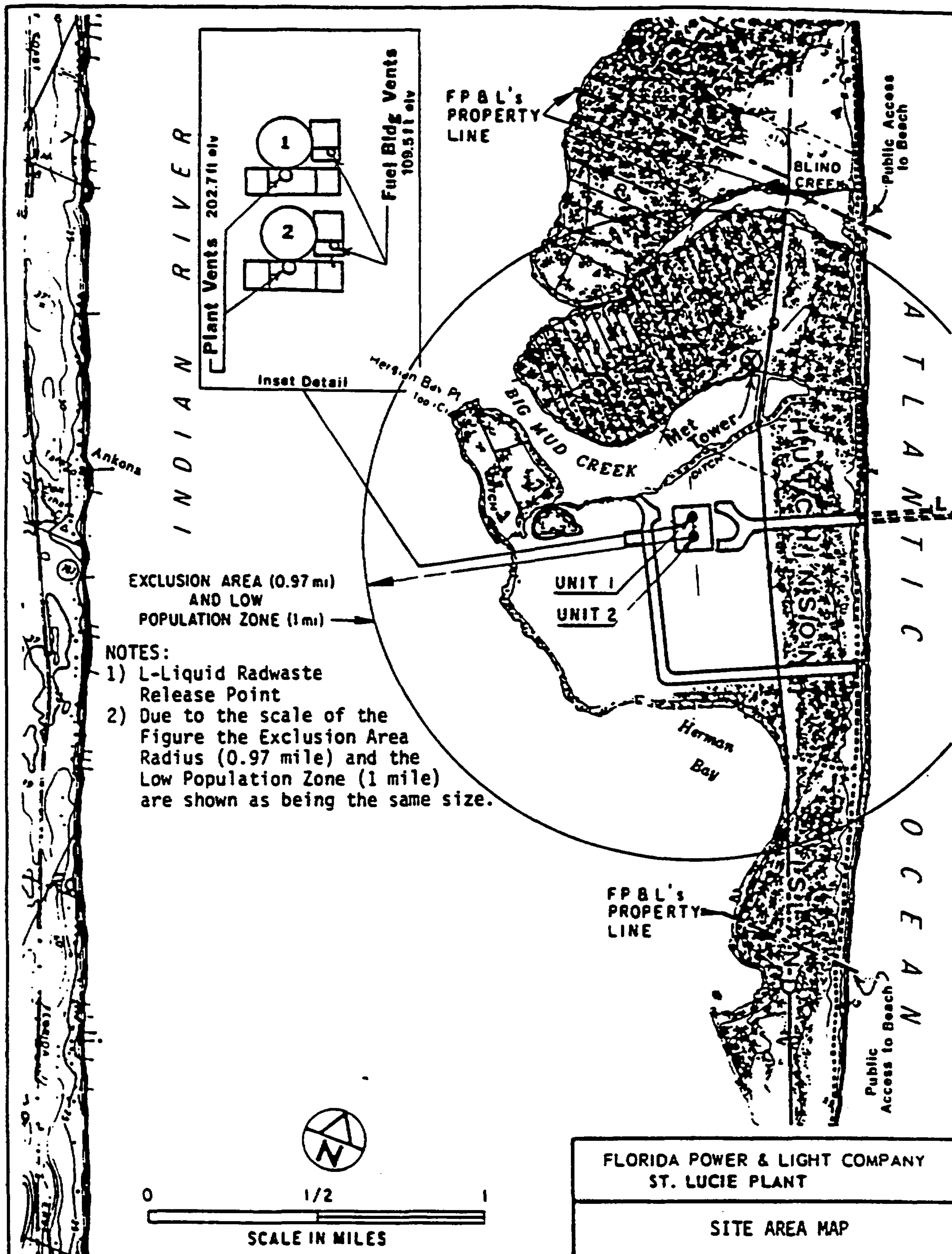
e. Cylinder wall minimum thickness = 3 feet.

f. Dome minimum thickness = 2.5 feet.

g. Dome inside radius = 112 feet.

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The steel reactor containment building is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

SITE AREA MAP

FIGURE 5.1-1

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing 236 fuel and poison rod locations. All fuel and poison rods are clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain approximately 1700 grams uranium. The initial core loading shall have a maximum enrichment of 2.73 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 91 full-length control element assemblies and no 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 to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 10,931 $\pm$ 275 cubic feet at a nominal T_{avg} of 572°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 a. The spent fuel pool and spent fuel storage racks shall be maintained with:

1. A k_{eff} equivalent to less than 1.0 when flooded with unborated water, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

2. A k_{eff} equivalent to less than or equal to 0.95 when flooded with water containing 520 ppm boron, including a conservative allowance for biases and uncertainties as described in Section 9.1 of the Updated Final Safety Analysis Report.

3. A nominal 8.96 inch center-to-center distance between fuel assemblies placed in the storage racks.

b. Fuel placed in Region I of the spent fuel storage racks shall be stored in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fresh fuel shall have a nominal average U-235 enrichment of less than or equal to 4.5 weight percent.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

4. The reactivity effects of fuel assembly burnup and decay time may be considered as specified in Figures 5.6-1c through 5.6-1e.

c. Fuel placed in Region II of the spent fuel storage racks shall be placed in a configuration that will assure compliance with 5.6.1 a.1 and 5.6.1 a.2, above, with the following considerations:

1. Fuel placed in Region II shall meet the burnup and decay time requirements specified in Figure 5.6-1a or 5.6-1b.

2. The reactivity effect of CEAs placed in fuel assemblies may be considered.

3. The reactivity equivalencing effects of burnable absorbers may be considered.

CRITICALITY (continued)

5.6.1 d. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a U-235 enrichment less than or equal to 4.5 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

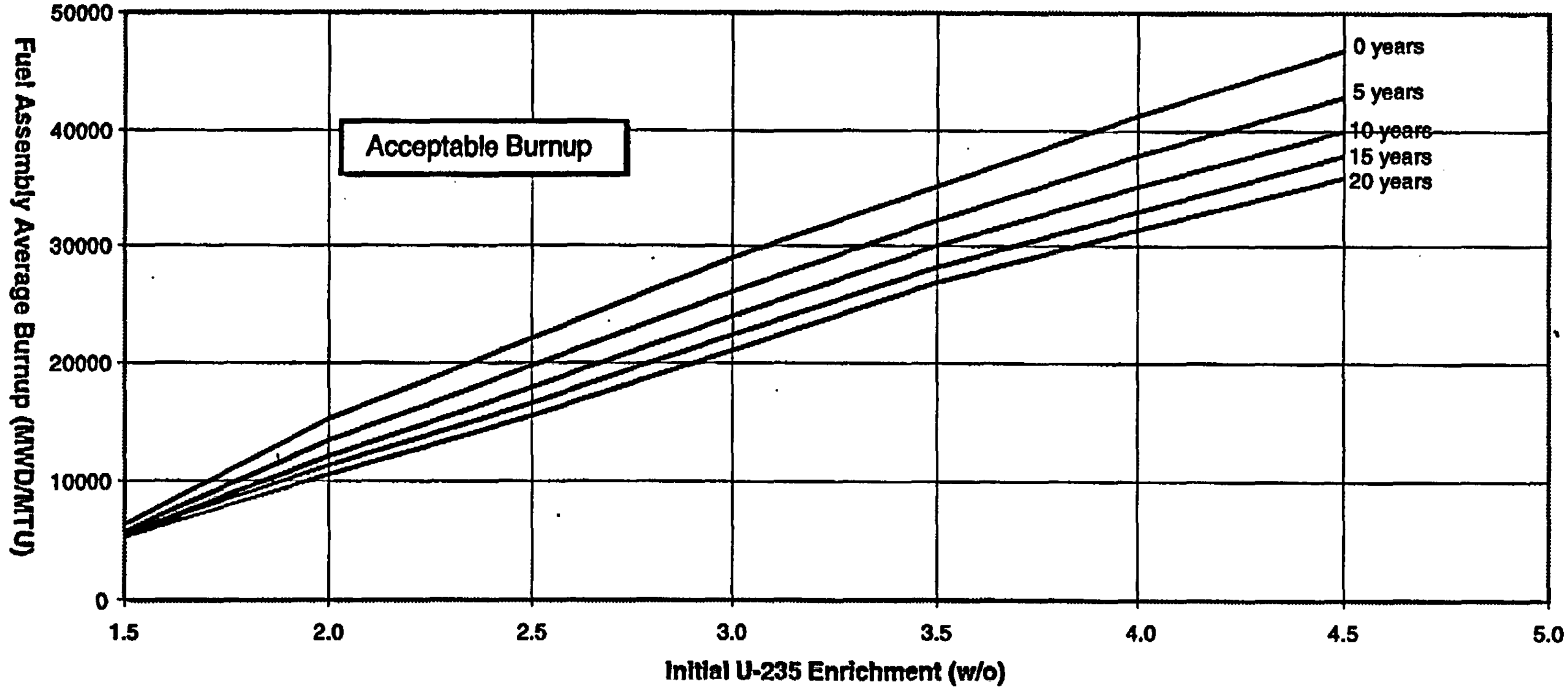
CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1360 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

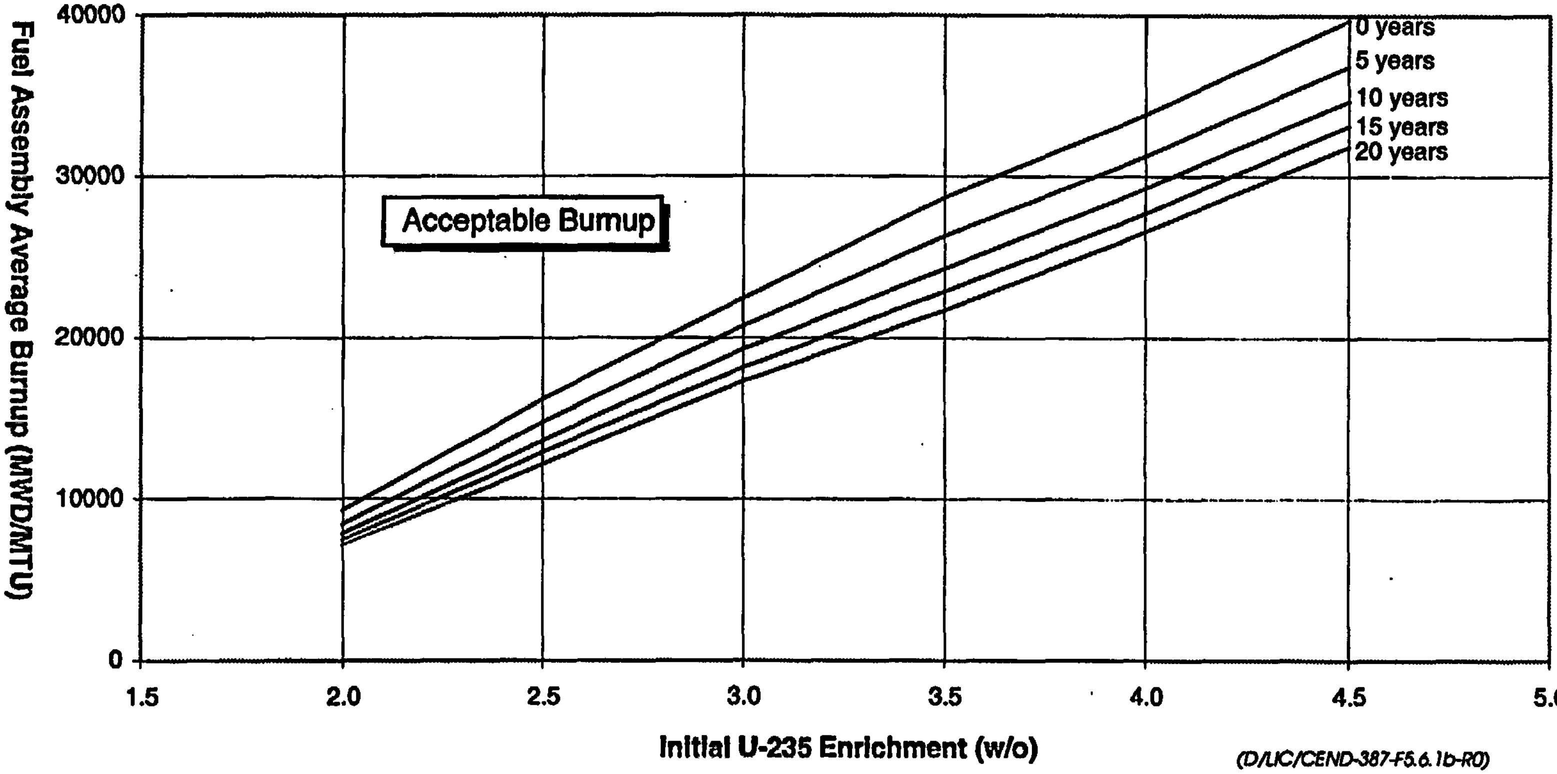
5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

Figure 5.6-1a
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region II, 1.3 w/o



(D/LIC/CEND-387-F5.6.1a-R0)

Figure 5.6-1b
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region II, 1.5 w/o



(D/UC/CEND-387-F5.6.1b-R0)

Figure 5.6-1c
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region I, 1.4 w/o

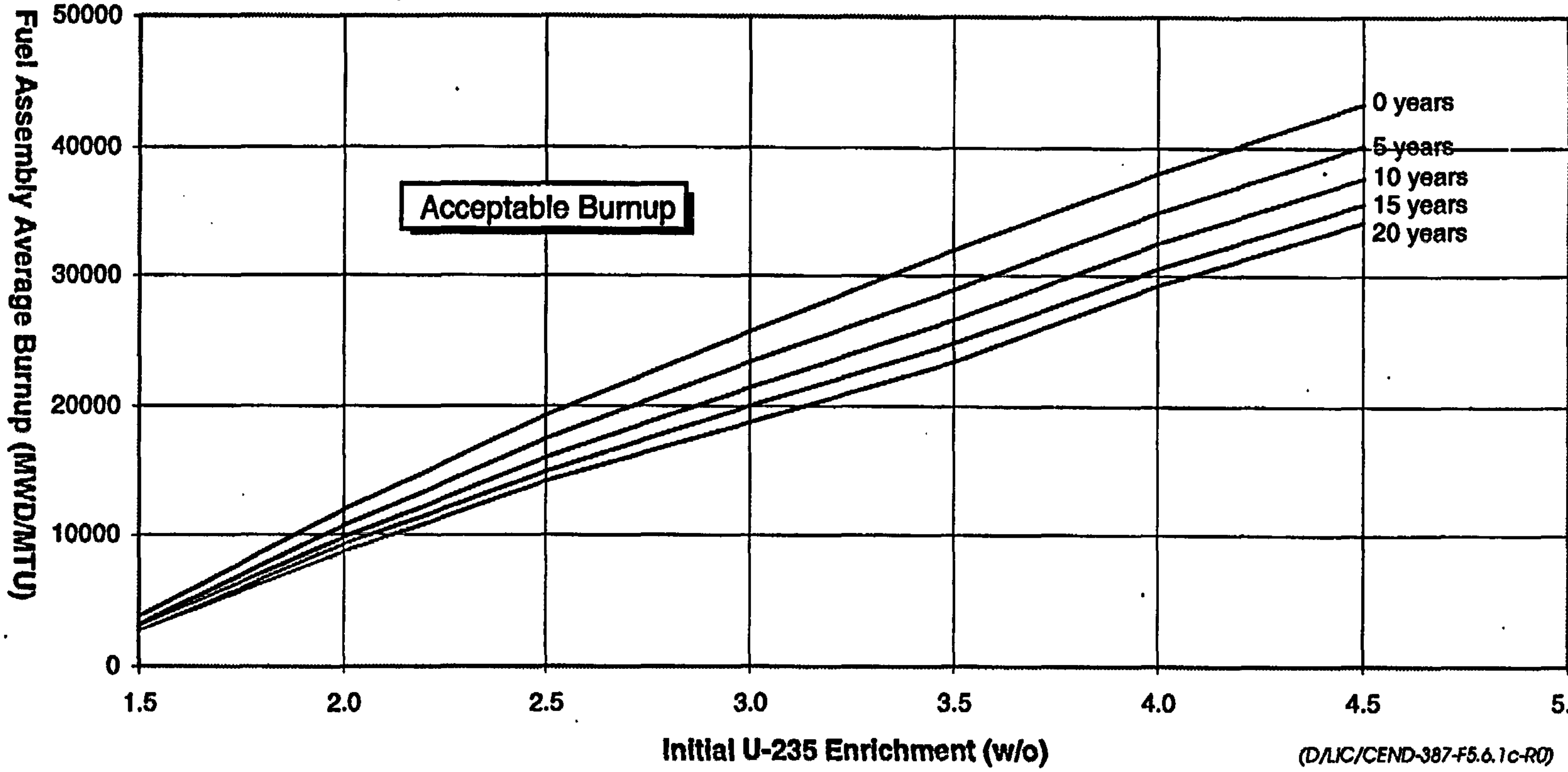
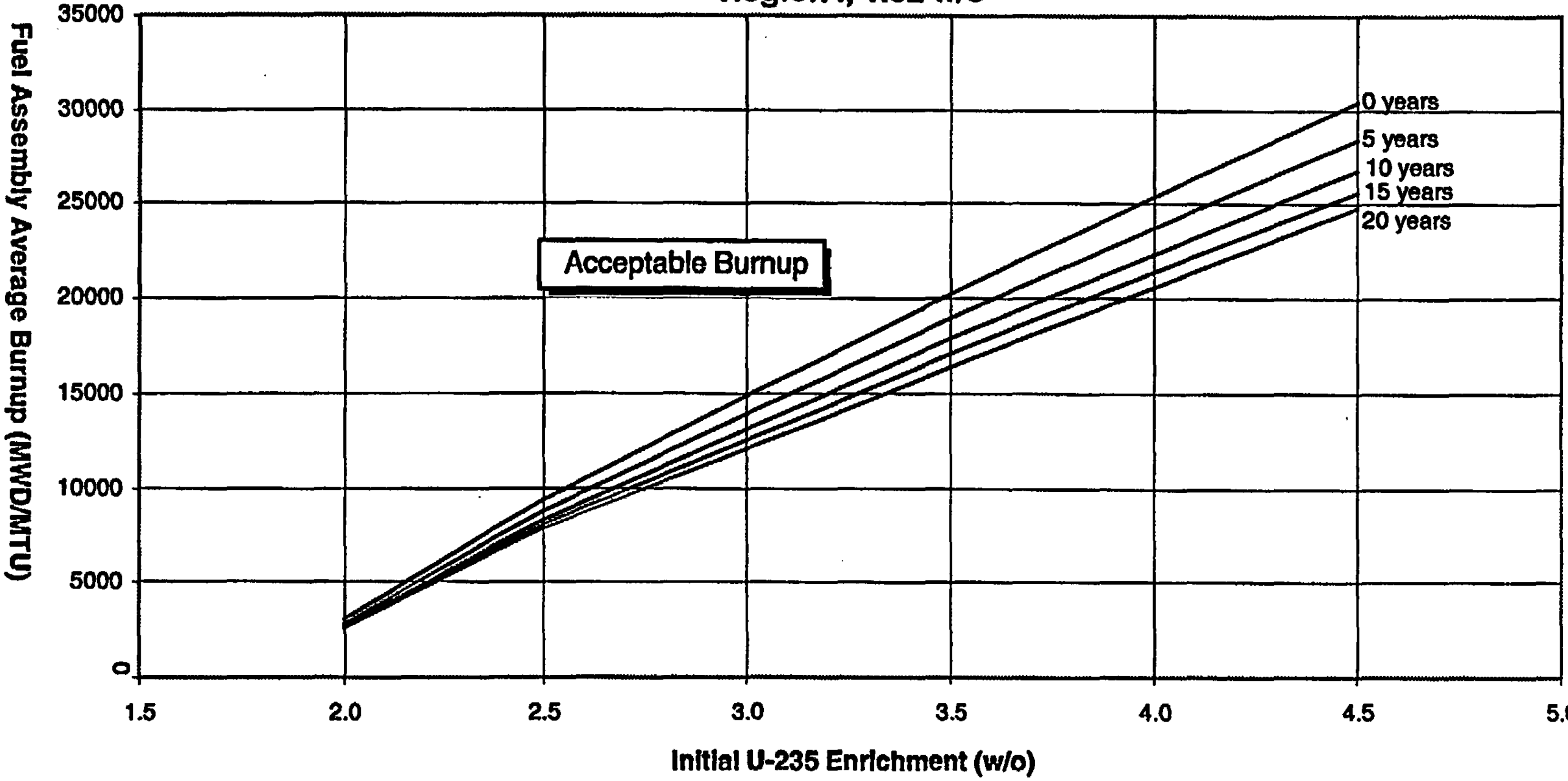


Figure 5.6-1d
Required Fuel Assembly Burnup vs Initial Enrichment and Decay Time
Region I, 1.82 w/o



(D/UC/CEND-387-F5.6.1d-R0)

Figure 5.6-1e
Required Fuel Assembly Burnup vs Initial Enrichment
Region I, 2.82 w/o

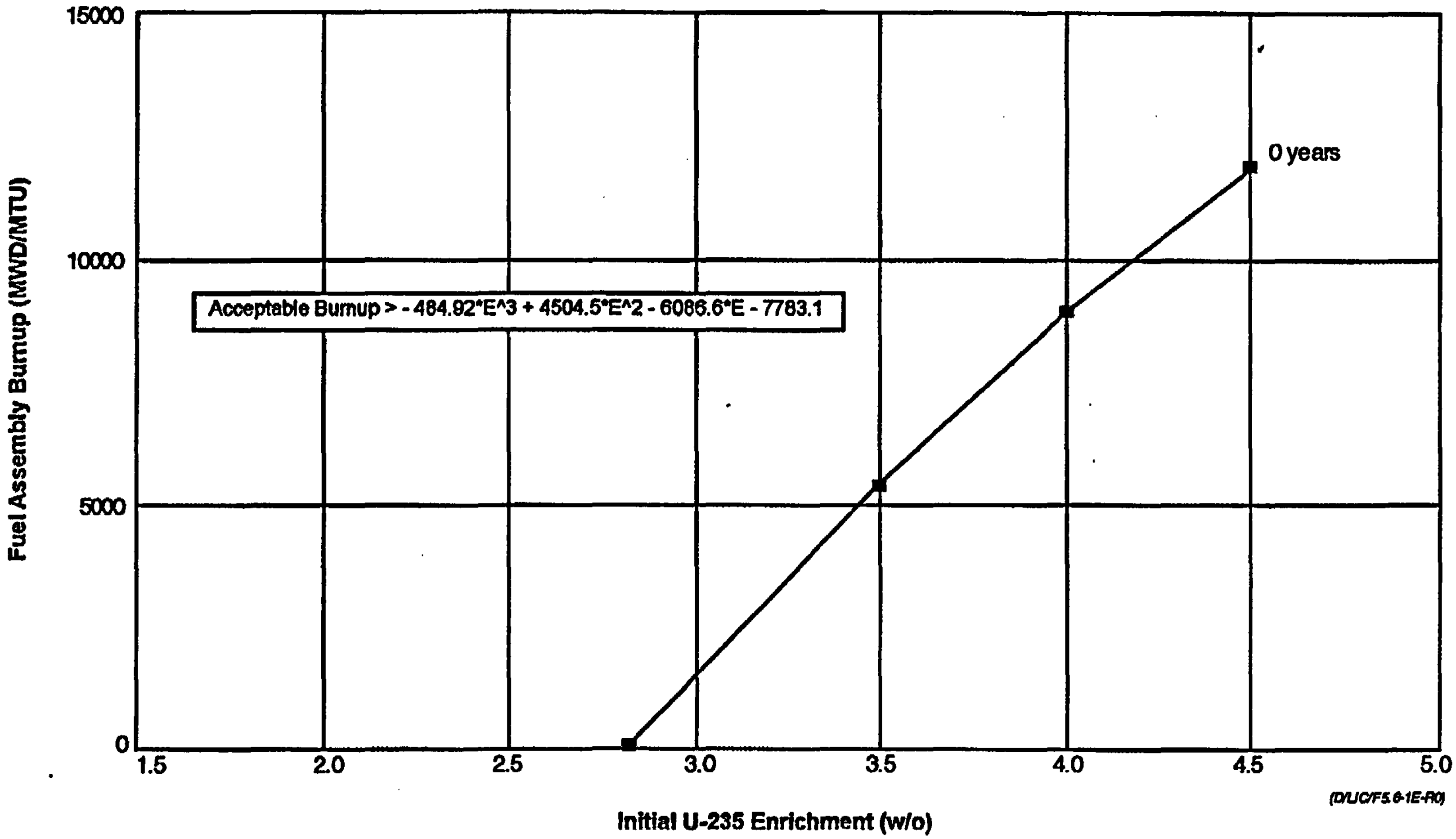


TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 545°F; cooldown cycle - T_{avg} from ≥ 545°F to ≤ 200°F.
	500 pressurizer heatup and cooldown cycles at rates ≤ 200°F/hr.	Heat cycle - Pressurizer temperature from < 200°F to > 653°F; cooldown ≥ 653°F to ≤ 200°F
	10 hydrostatic testing cycles.	RCS pressurized to 3110 psig with RCS temperature ≥ 60°F above the most limiting components' NDTT value.
	200 leak testing cycles.	RCS pressured to 2250 psia with RCS temperature greater than minimum for hydrostatic testing, but less than minimum RCS temperature for critically.
	400 reactor trip cycles.	Trip from 100% of RATED THERMAL POWER.
	40 turbine trip cycles with delayed reactor trip.	Turbine trip (total load rejection) from 100% of RATED THERMAL POWER. followed by resulting reactor trip.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	40 complete loss of reactor coolant flow cycles.	Simultaneous loss of all Reactor Coolant Pumps at 100% of RATED THERMAL POWER.
	5 complete loss of secondary pressure cycles.	Loss of secondary pressure from either steam generator while in MODE 1, 2 or 3.
	100 pressurizer spray cycles per year with pressurizer/spray water $\Delta T > 200°F$ or as otherwise calculated by the following method:	Spray operation consisting of opening and closing either the main or auxiliary spray valves(s) spray water/pressurizer $\Delta T > 200°F$.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Method for Calculating Pressurizer Spray Nozzle Cumulative Usage Factor

ΔT	N_A	N	N/N_A
201 - 300	13,000		
301 - 400	5,000		
401 - 500	3,000		
501 - 600	1,500		
		$\Sigma N/N_A$	

Where:

ΔT = Temperature difference between pressurizer water and spray in °F.

N_A = Allowable number of spray cycles.

N = Number of cycles in ΔT range indicated.

TABLE 5.7-1 (Continued)

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System		

Calculational Method:

1. At 12-month intervals the cumulative spray cycles shall be totaled. If the total is equal to or less than 1000, no further action is required.

2. If the cumulative total exceeds 1000, the spray nozzle usage factor shall be calculated as follows:

   A. Fill in Column "N" above.

   B. Calculate "N/N_A" (Divide N and N_A).

   C. Add Column "N/N_A" to find $\Sigma N/N_A$.

   $\Sigma N/N_A$ is the cumulative spray nozzle usage factor. If the calculated usage factor is equal to or less than 0.75, no further action is required.

3. If the calculated usage factor exceeds 0.75, subsequent pressurizer spray operation shall be restricted so that the difference between the pressurizer water temperature and the spray water temperature shall be limited to less than or equal to 200°F when spray is operated. An engineering evaluation of nozzle fatigue shall be performed and shall determine that that the nozzle remains acceptable for additional service prior to removing this restriction.

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant General 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 Shift Supervisor, or during his absence from the control room, a designated individual, shall be responsible for the control room command function. A management directive to this effect, signed by the Site Vice President, shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

ONSITE AND OFFSITE ORGANIZATION

6.2.1 An onsite and an offsite organization shall be established for unit operation and corporate management. This onsite and offsite organization shall include the positions for activities affecting the safety of the nuclear power plant.

a. Lines of authority, responsibility and communication shall be established and defined from the highest management levels through intermediate levels to and including all operating organization positions. Those relationships shall be documented and updated, as appropriate, in the form of organizational charts. These organizational charts will be documented in the Topical Quality Assurance Report and updated in accordance with 10 CFR 50.54(a)(3).

b. The Chief Nuclear Officer shall be responsible for overall plant nuclear safety. This individual shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support in the plant so that continued nuclear safety is assured.

c. The Plant General Manager shall be responsible for overall unit safe operation and shall have control over those onsite resources necessary for safe operation and maintenance of the plant.

d. Although the individuals who train the operating staff and those who carry out the quality assurance functions may report to the appropriate manager onsite, they shall have sufficient organizational freedom to be independent from operating pressures.

e. Although health physics individuals may report to any appropriate manager onsite, for matters relating to radiological health and safety of employees and the public, the Health Physics Supervisor shall have direct access to that onsite individual having responsibility for overall unit management. Health physics personnel shall have the authority to cease any work activity when worker safety is jeopardized or in the event of unnecessary personnel radiation exposures.

6.2 ORGANIZATION (Continued)

UNIT STAFF

6.2.2 The unit organization shall be subject to the following:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the reactor is in MODE 1, 2, 3, or 4, at least one licensed Senior Reactor Operator shall be in the control room.

c. A health physics technician# shall be on site when fuel is in the reactor.

d. Either a licensed SRO or licensed SRO limited to fuel handling who has no concurrent responsibilities during this operation shall be present during fuel handling and shall directly supervise all CORE ALTERATIONS.

e. DELETED

f. Administrative procedures shall be developed and implemented to limit the working hours of personnel who perform safety-related functions (e.g., licensed senior reactor operators (SROs), licensed reactor operators (ROs), health physicists, auxiliary operators, and key maintenance personnel). The administrative procedures shall include guidelines on working hours that ensure that adequate shift coverage shall be maintained without routine heavy use of overtime.

Any deviation from the above guidelines shall be authorized by the Plant General Manager or the Plant General Manager's designee, in accordance with approved administrative procedures, and with documentation of the basis for granting the deviation. Controls shall be included in the procedures to require a periodic independent review be conducted to ensure that excessive hours have not been assigned. Routine deviation from the working hour guidelines shall not be authorized.

g. The Operations Supervisor shall hold a Senior Reactor Operator License.

The health physics technician may be less than the minimum requirement 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.

DELETED

DELETED

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION
TWO UNITS WITH TWO SEPARATE CONTROL ROOMS

WITH UNIT 1 IN MODE 5 OR 6 OR DEFUELED		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS (SRO)	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	2[b]
STA *	1	None

WITH UNIT 1 IN MODE 1, 2, 3 or 4		
POSITION	NUMBER OF INDIVIDUALS REQUIRED TO FILL POSITION	
	MODE 1, 2, 3, or 4	MODE 5 or 6
SS (SRO)	1[a]	1[a]
SRO	1	None
RO	2	1
AO	2	1
STA *	1[c]	None

SS - Shift Supervisor with a Senior Reactor Operator's License on Unit 2
SRO - Individual with a Senior Reactor Operator's License on Unit 2
RO - Individual with a Reactor Operator's License on Unit 2
AO - Auxiliary Operator
STA - Shift Technical Advisor

Except for the Shift Supervisor, the Shift Crew Composition may be one less than the minimum requirements of Table 6.2-1 for a period of time not to exceed 2 hours in order to accommodate unexpected absence of on-duty shift crew members provided immediate action is taken to restore the Shift Crew Composition to within the minimum requirements of Table 6.2-1. This provision does not permit any shift crew position to be unmanned upon shift change due to an oncoming shift crewman being late or absent.

During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 1, 2, 3 or 4, an individual (other than the Shift Technical Advisor) with a valid SRO license shall be designated to assume the Control Room command function. During any absence of the Shift Supervisor from the Control Room while the unit is in MODE 5 or 6, an individual with a valid SRO or RO license shall be designated to assume the Control Room command function.

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a/ Individual may fill the same position on Unit 1.

b/ One of the two required individuals may fill the same position on Unit 1.

c/ If STA position is filled by an STA qualified Shift Supervisor or dedicated STA, then the individual may fill the same position on Unit 1.

* A single, onsite STA position shall be manned in Mode 1, 2, 3, and 4 unless the Shift Supervisor meets the qualifications for the STA as required by Technical Specification 6.3.1 or an individual on each unit with a Senior Reactor Operator's license meets the qualifications for the STA as required by Technical Specification 6.3.1.

6.2.3 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor function is to provide on shift advisory technical support in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI / ANS-3.1-1978 for comparable positions, except for:

(1) the Health Physics Supervisor who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975,

(2) the Shift Technical Advisor who shall have specific training in plant design and plant operating characteristics, including transients and accidents, and any of the following educational requirements:

- Bachelor's degree in engineering from an accredited institution; or
- Professional Engineer's (PE) license obtained by successful completion of the PE examination; or
- Bachelor's degree in engineering technology from an accredited institution, including course work in the physical, mathematical, or engineering sciences, or
- Bachelor's degree in physical science from an accredited institution, including course work in the physical, mathematical, or engineering sciences.

(3) the Multi-Discipline Supervisors who shall meet or exceed the following requirements:

a. Education: Minimum of a high school diploma or equivalent.

b. Experience: Minimum of four years of related technical experience, which shall include three years power plant experience of which one year is at a nuclear power plant.

c. Training: Complete the Multi-Discipline Supervisor training program.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI / ANS-3.1-1978 and 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.5 REVIEW AND AUDIT

6.5.1 FACILITY REVIEW GROUP (FRG)

FUNCTION

6.5.1.1 The Facility Review Group shall function to advise the Plant General Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The FRG shall have voting members composed of individuals from each of the following disciplines:

Operations	Electrical Maintenance
Reactor Engineering	Mechanical Maintenance
Health Physics	Technical Support
Chemistry	Quality Assurance / Control
Licensing	Services
Instrument and Control	

The Plant General Manager shall appoint the FRG members, in writing, and from this membership shall designate, in writing, an FRG Chairman.

Members shall meet or exceed the qualifications required for Managers, Supervisors, or Professional-Technical, as appropriate, pursuant to Specification 6.3.1.

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the FRG Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in FRG activities at any one time.

MEETING FREQUENCY

6.5.1.4 The FRG shall meet at least once per calendar month and as convened by the FRG Chairman or his designated alternate.

QUORUM

6.5.1.5 The quorum of the FRG necessary for the performance of the FRG 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 Facility Review Group shall be responsible for:

a. Review of (1) all new procedures required by Specification 6.8 and all procedure changes that require a written 50.59 evaluation, (2) all programs required by Specification 6.8 and changes thereto, and (3) any other proposed procedures or changes thereto as determined by the Plant General 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 Chief Nuclear Officer and to the Chairman of the Company Nuclear Review Board.

f. Review of all REPORTABLE EVENTS.

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 General Manager or the Company Nuclear Review Board.

i. Not Used.

j. Not Used.

RESPONSIBILITIES (Continued)

k. Review of every unplanned on-site release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Chief Nuclear Officer and to the Company Nuclear Review Board.

l. Review of changes to the PROCESS CONTROL PROGRAM and the OFFSITE DOSE CALCULATION MANUAL and RADWASTE TREATMENT SYSTEMS.

m. Review and documentation of judgment concerning prolonged operation in bypass, channel trip, and/or repair of defective protection channels of process variables placed in bypass since the last FRG meeting.

n. Review of the Fire Protection Program and implementing procedures and submittal of recommended changes to the Company Nuclear Review Board.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend in writing to the Plant General Manager approval or disapproval of items considered under Specifications 6.5.1.6a. through d. and m. above.

b. Render determinations in writing with regard to whether or not each item considered under Specifications 6.5.1.6a, b, d, and e above requires NRC approval pursuant to 10 CFR 50.59.

c. Provide written notification within 24 hours to the Chief Nuclear Officer and the Company Nuclear Review Board of disagreement between the FRG and the Plant General Manager; however, the Plant General Manager shall have responsibility for resolution of such disagreements pursuant to Specification 6.1.1 above.

RECORDS

6.5.1.8 The Facility Review Group shall maintain written minutes of each FRG meeting that, at a minimum, document the results of all FRG activities performed under the responsibility and authority provisions of these technical specifications. Copies shall be provided to the Plant General Manager, Chief Nuclear Officer and the Chairman of the Company Nuclear Review Board.

6.5.2 COMPANY NUCLEAR REVIEW BOARD (CNRB)

FUNCTION

6.5.2.1 The Company Nuclear Review Board 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

FUNCTION (Continued)

e. instrument and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The Chief Nuclear Officer shall appoint, in writing, a minimum of five members to the CNRB and shall designate from this membership, in writing, a Chairman. The membership shall function to provide independent review and audit in the areas listed in Specification 6.5.2.1. The Chairman shall meet the requirements of ANSI/ANS-3.1-1987, Section 4.7.1. The members of the CNRB shall meet the educational requirements of the ANSI/ANS-3.1-1987, Section 4.7.2, and have at least 5 years of professional level experience in one or more of the fields listed in Specification 6.5.2.1. CNRB members who do not possess the educational requirements of ANSI/ANS-3.1-1987, Section 4.7.2 (up to a maximum of 2 members) shall be evaluated, and have their membership approved and documented, in writing, on a case-by-base basis by the Chief Nuclear Officer, considering the alternatives to educational requirements of ANSI/ANS-3.1-1987, Sections 4.1.1 and 4.1.2.

ALTERNATES

6.5.2.3 All alternate members shall be appointed in writing by the CNRB Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in CNRB activities at any one time.

CONSULTANTS

6.5.2.4 Consultants shall be utilized as determined by the CNRB Chairman to provide expert advice to the CNRB.

MEETING FREQUENCY

6.5.2.5 The CNRB shall meet at least once per calendar quarter during the initial year of unit operation following fuel loading and at least once per 6 months thereafter and as convened by the CNRB Chairman or his designated alternate.

QUORUM

6.5.2.6 The quorum of the CNRB necessary for the performance of the CNRB review and audit functions of these Technical Specifications shall consist of the Chairman or his designated alternate and at least a majority of CNRB members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the unit.

REVIEW

6.5.2.7 The CNRB shall review:

a. The evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments completed under the provisions of Section 50.59, 10 CFR, to verify that such actions did not require NRC approval pursuant to 10 CFR 50.59.

b. Proposed changes to procedures, equipment, or systems which require NRC approval pursuant to 10 CFR 50.59.

c. Proposed tests or experiments which require NRC approval pursuant to 10 CFR 50.59.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. All REPORTABLE EVENTS.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meeting minutes of the Facility Review Group.

AUDITS

6.5.2.8 Audits of unit activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of unit operation to provisions contained within the Technical Specifications and applicable license conditions.

b. The performance, training and qualifications of the entire unit staff.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety.

AUDITS (continued)

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50.

e. Any other area of unit operation considered appropriate by the CNRB or the Chief Nuclear Officer.

f. The fire protection programmatic controls including the implementing procedures at least once per 24 months by qualified licensee QA personnel.

g. The fire protection equipment and program implementation at least once per 12 months utilizing either a qualified offsite licensee fire protection engineer or an outside independent fire protection consultant. An outside independent fire protection consultant shall be used at least every third year.

h. The radiological environmental monitoring program and the results thereof.

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures.

j. The PROCESS CONTROL PROGRAM and implementing procedures for dewatering of radioactive bead resin.

TECHNICAL REVIEW RESPONSIBILITIES

6.5.2.9 The technical review responsibilities under the cognizance of the CNRB shall encompass:

a. Plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources that may indicate areas for improving plant safety;

b. Plant operations, modifications, maintenance, and surveillance to verify independently that these activities are performed safely and correctly and that human errors are reduced as much as practical;

c. Internal and external operational experience information that may indicate areas for improving plant safety; and

d. Making detailed recommendations to the Chairman – CNRB and plant management for revising procedures, equipment modifications or other means of improving nuclear safety and plant reliability.

AUTHORITY

6.5.2.10 The CNRB shall report to and advise the Chief Nuclear Officer on those areas of responsibility specified in Specifications 6.5.2.7, 6.5.2.8, and 6.5.2.9.

RECORDS

6.5.2.11 Records of CNRB activities shall be prepared, approved, and distributed as indicated below:

a. Minutes of each CNRB meeting shall be prepared, approved, and forwarded to the Chief Nuclear Officer within 14 days following each meeting.

b. Reports of reviews encompassed by Specification 6.5.2.7 above shall be prepared, approved, and forwarded to the Chief Nuclear Officer within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 above shall be forwarded to the Chief Nuclear Officer and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

d. Technical reviews encompassed by Specification 6.5.2.9 above shall be prepared, maintained and a report of the activities forwarded each calendar month to the Chairman, CNRB.

6.6 REPORTABLE EVENTS ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the FRG, and the results of this review shall be submitted to the CNRB, and the Chief Nuclear Officer.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Chief Nuclear Officer and the CNRB shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the FRG. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB, and the Chief Nuclear Officer within 14 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES AND PROGRAMS

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978, and those required for implementing the requirements of NUREG 0737.

b. Refueling operations.

c. Surveillance and test activities of safety-related equipment.

d. Not Used.

e. Not Used.

PROCEDURES AND PROGRAMS (Continued)

f. Fire Protection Program implementation.

g. PROCESS CONTROL PROGRAM implementation.

h. OFFSITE DOSE CALCULATION MANUAL implementation.

i. Quality Control Program for effluent monitoring, using the guidance in Regulatory Guide 1.21, Revision 1, June 1974.

j. Quality Control Program for environmental monitoring using the guidance in Regulatory Guide 4.1, Revision 1, April 1975.

6.8.2 REVIEW AND APPROVAL OF PROCEDURES:

Each new procedure of Specification 6.8.1a. through i. above shall be independently reviewed by an individual or group from the appropriate discipline(s), and shall be reviewed by the FRG. New procedures shall be approved by the Plant General Manger or individuals designated in writing by the Plant General Manager prior to implementation. Each procedure of Specification 6.8.1 shall be reviewed periodically as set forth in administrative procedures.

6.8.3 CHANGES TO PROCEDURES:

a. Each revision to the procedures of Specification 6.8.1a. through i. above shall be independently reviewed by an individual or group from the appropriate discipline(s), and revisions that require a written evaluation pursuant to 10 CFR 50.59 shall be reviewed by the FRG. Procedure revisions shall be approved by the Plant General Manager or individuals designated in writing by the Plant General Manager prior to implementation.

b. Temporary changes to procedures of Specification 6.8.1a. through i. above may be made provided:

   1. The intent of the original procedure is not altered.

   2. 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.

   3. The change is documented and, if appropriate, incorporated in the next revision of the affected procedure pursuant to Specification 6.8.3.a.

PROCEDURES AND PROGRAMS (Continued)

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the CNRB:

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 Shutdown Cooling System, High Pressure Safety Injection System, Containment Spray System, and RCS Sampling. The program shall include the following:

(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 Radioiodine 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.

PROCEDURES AND PROGRAMS (continued)

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 of these variables,

(ii) Identification of the procedures used to measure the values of the critical variables,

(iii) Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in-leakage,

(iv) Procedures for the recording and management of data,

(v) Procedures defining corrective actions for all off-control point chemistry conditions, and

(vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

(i) Training of personnel, and

(ii) Procedures for monitoring.

e. DELETED

f. Radioactive Effluent Controls Program

A program shall be provided conforming with 10 CFR 50.36a for the control of radioactive effluents and for maintaining the doses to MEMBERS OF THE PUBLIC from radioactive effluents as low as reasonably achievable. The program (1) shall be contained in the ODCM, (2) shall be implemented by operating procedures, and (3) shall include remedial actions to be taken whenever the program limits are exceeded. The program shall include the following elements:

1) Limitations on the operability of radioactive liquid and gaseous monitoring instrumentation including surveillance tests and setpoint determination in accordance with the methodology in the ODCM,

2) Limitations on the concentrations of radioactive material released in liquid effluents to UNRESTRICTED AREAS conforming to ten times the concentration values in 10 CFR 20.1001 - 20.2401, Appendix B, Table 2, Column 2.

3) Monitoring, sampling, and analysis of radioactive liquid and gaseous effluents in accordance with 10 CFR 20.1302 and with the methodology and parameters in the ODCM,

4) Limitations on the annual and quarterly doses or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released from each unit to UNRESTRICTED AREAS conforming to Appendix I to 10 CFR Part 50,

5) Determination of cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with the methodology and parameters in the ODCM at least every 31 days,

6) Limitations on the operability and use of the liquid and gaseous effluent treatment systems to ensure that the appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a 31-day period would exceed 2 percent of the guidelines for the annual dose or dose commitment conforming to Appendix I to 10 CFR Part 50,

7) Limitations on the dose rate resulting from radioactive material released in gaseous effluents to areas at or beyond the SITE BOUNDARY 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 Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ;

8) Limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

9) Limitations on the annual and quarterly doses to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium, and all radionuclides in particulate form with half-lives greater

than 8 days in gaseous effluents released from each unit to areas beyond the SITE BOUNDARY conforming to Appendix I to 10 CFR Part 50,

10) Limitations on the annual dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources conforming to 40 CFR Part 190.

g. Radiological Environmental Monitoring Program

A program shall be provided to monitor the radiation and radionuclides in the environs of the plant. The program shall provide (1) representative measurements of radioactivity in the highest potential exposure pathways, and (2) verification of the accuracy of the effluent monitoring program and modeling of the environmental exposure pathways. The program shall (1) be contained in the ODCM, (2) conform to the guidance of Appendix I to 10 CFR Part 50, and (3) include the following:

1) Monitoring, sampling, analysis, and reporting of radiation and radionuclides in the environment in accordance with the methodology and parameters in the ODCM.

2) A Land Use Census to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the monitoring program are made if required by the results of this census, and

3) Participation in a Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring.

h. Containment Leakage Rate Testing Program

A program to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50 Appendix J, Option B, as modified by approved exemptions. This program is in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," as modified by the following exception(s):

a) Bechtel Topical Report, BN-TOP-1 or ANS 56.8-1994 (as recommended by R.G. 1.163) will be used for type A testing.

b) The first Type A test performed after the June 1992 Type A test shall be no later than June 2007.

The peak calculated containment internal pressure for the design basis loss of coolant accident P_a, is 41.8 psig. The containment design pressure is 44 psig.

The maximum allow containment leakage rate, L_a, at P_a, shall be 0.50% of containment air weight per day.

Leakage rate acceptance criteria:

a. Containment leakage rate acceptance criterion is $\leq 1.0\ L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60\ L_a$ for the Type B and C tests, $\leq 0.75\ L_a$ for Type A tests, and $\leq 0.12\ L_a$ for secondary containment bypass leakage paths.

b. Air lock testing acceptance criteria are:

1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

2. For each door seal, leakage rate is $< 0.01\ L_a$ when pressurized to $\geq P_a$.

The provisions of T.S. 4.0.2 do not apply to test frequencies in the Containment Leak Rate Testing Program.

The provisions of T.S. 4.0.3 are applicable to the Containment Leak Rate Testing Program.

i. Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2 and 3 components (pumps and valves). The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code* and applicable addenda as follows:

ASME Boiler and Pressure Vessel Code* and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice testing activities;

c. The provisions of Specification 4.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code* shall be construed to supersede the requirements of any technical specification.

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* Where ASME Boiler and Pressure Vessel Code is referenced it also refers to the applicable portions of ASME/ANSI OM-Code, "Operation and Maintenance of Nuclear Power Plants," with applicable addenda, to the extent it is referenced in the Code.

j. Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

1. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

2. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:

   a. a change in the TS incorporated in the license; or

   b. a change to the updated UFSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.

3. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the UFSAR.

4. Proposed changes that meet the criteria of Specification 6.8.4.j.2.a or 6.8.4.j.2.b, above, shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) for whom monitoring was required receiving Annual Deep Dose Equivalent exposures

[1/] A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

ANNUAL REPORTS (Continued)

greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,[2/] e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total Deep Dose Equivalent received from external sources should be assigned to specific major work functions.

b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC, no later than the 15th of each month following the calendar month covered by the report.

[2/] This tabulation supplements the requirements of 20.2206 of 10 CFR Part 20.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.7 The Annual Radioactive Effluent Release Report covering the operation of the unit during the previous 12 months of operation shall be submitted within 60 days after January 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR Part 50.

*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.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT**

6.9.1.8 The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted before May 1 of each year. The report shall include summaries, interpretations, and analysis of trends of the results of the Radiological Environmental Monitoring Program for the reporting period. The material provided shall be consistent with the objectives outlined in (1) the ODCM and (2) Sections IV.B.2, IV.B.3, and IV.C of Appendix I to 10 CFR Part 50.

**A single submittal may be made for a multiple unit station.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (continued)

6.9.1.9 At least once every 5 years, an estimate of the actual population within 10 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.10 At least once every 10 years, an estimate of the actual population within 50 miles of the plant shall be prepared and submitted to the NRC.

6.9.1.11 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Specification 3.1.1.1	Shutdown Margin – T_{avg} Greater than 200°F
Specification 3.1.1.2	Shutdown Margin – T_{avg} Less Than or Equal to 200°F
Specification 3.1.1.4	Moderator Temperature Coefficient
Specification 3.1.3.1	Movable Control Assemblies – CEA Position
Specification 3.1.3.6	Regulating CEA Insertion Limits
Specification 3.2.1	Linear Heat Rate
Specification 3.2.2	Total Planar Radial Peaking Factors - F^T_{XY}
Specification 3.2.3	Total Integrated Radial Peaking Factors – F_r^T
Specification 3.2.5	DNB Parameters – Axial Shape Index
Specification 3.9.1	Refueling Operations – Boron Concentration

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, as described in the following documents or any approved Revisions and Supplements thereto:

1. WCAP-11596-P-A, "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores," June 1988 (Westinghouse Proprietary).

2. NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants," Florida Power & Light Company, January 1995.

3. CENPD-199-P, Rev. 1-P-A, "C-E Setpoint Methodology: CE Local Power Density and DNB LSSS and LCO Setpoint Methodology for Analog Protection Systems," January 1986.

4. CENPD-266-P-A, "The ROCS and DIT Computer Code for Nuclear Design," April 1983.

5. CENPD-275-P, Revision 1-P-A, "C-E Methodology for Core Designs Containing Gadolinia-Urania Burnable Absorbers," May 1988.

6. CENPD-188-A, "HERMITE: A Multi-Dimensional Space – Time Kinetics Code for PWR Transients," July 1976.

CORE OPERATING LIMITS REPORT (COLR) (Continued)

b. (Continued)

7. CENPD-153-P, Rev. 1-P-A, "Evaluation of Uncertainty in the Nuclear Power Peaking Measured by the Self-Powered, Fixed Incore Detector System," May 1980.

8. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 1: CE Calculated Local Power Density and Thermal Margin/Low Pressure LSSS for St. Lucie Unit 1," December 1979.

9. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 2: Combination of System Parameter Uncertainties in Thermal Margin Analyses for St. Lucie Unit 1," January 1980.

10. CEN-123(F)-P, "Statistical Combination of Uncertainties Methodology Part 3: CE Calculated Departure from Nucleate Boiling and Linear Heat Rate Limiting Conditions for Operation for St. Lucie Unit 1," February 1980.

11. CEN-191(B)-P, "CETOP-D Code Structure and Modeling Methods for Calvert Cliffs Units 1 and 2," December 1981.

12. Letter, J.W. Miller (NRC) to J.R. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval of CEN-123(F)-P (three parts) and CEN-191(B)-P).

13. CEN-371(F)-P, "Extended Statistical Combination of Uncertainties," July 1989.

14. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL), Docket No. 50-389, "St. Lucie Unit 2 - Change to Technical Specification Bases Sections '2.1.1 Reactor Core' and '3/4.2.5 DNB Parameters' (TAC No. M87722)," March 14, 1994 (Approval of CEN-371(F)-P).

15. CENPD-161-P-A, "TORC Code, A Computer Code for Determining the Thermal Margin of a Reactor Core," April 1986.

16. CENPD-162-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 1, Uniform Axial Power Distribution," April 1975.

17. CENPD-207-P-A, "Critical Heat Flux Correlation for C-E Fuel Assemblies with Standard Spacer Grids Part 2, Non-uniform Axial Power Distribution," December 1984.

18. CENPD-206-P-A, "TORC Code, Verification and Simplified Modeling Methods," June 1981.

CORE OPERATING LIMITS REPORT (COLR) (Continued)

b. (Continued)

19. CENPD-225-P-A, "Fuel and Poison Rod Bowing," June 1983.

20. CENPD-139-P-A, "C-E Fuel Evaluation Model Topical Report," July 1974.

21. CEN-161(B)-P-A, "Improvements to Fuel Evaluation Model," August 1989.

22. CEN-161(B)-P, Supplement 1-P-A, "Improvements to Fuel Evaluation Model," January 1992.

23. CENPD-132, Supplement 3-P-A, "Calculative Methods for the C-E Large Break LOCA Evaluation Model for the Analysis of C-E and W Designed NSSS," June 1985.

24. CENPD-133, Supplement 5-A, "CEFLASH-4A, A FORTRAN77 Digital Computer Program for Reactor Blowdown Analysis," June 1985.

25. CENPD-134, Supplement 2-A, "COMPERC-II, a Program for Emergency Refill-Reflood of the Core," June 1985.

26. CENPD-135-P, Supplement 5, "STRIKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.

27. Letter, R.L. Baer (NRC) to A.E. Scherer (CE), "Evaluation of Topical Report CENPD-135, Supplement #5," September 6, 1978.

28. CENPD-137, Supplement 1-P, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," January 1977.

29. CENPD-133, Supplement 3-P, "CEFLASH-4AS, A Computer Program for the Reactor Blowdown Analysis of the Small Break Loss of Coolant Accident," January 1977.

30. Letter, K. Kniel (NRC) to A.E. Scherer (CE), "Evaluation of Topical Reports CENPD-133, Supplement 3-P and CENPD-137, Supplement 1-P," September 27, 1977.

31. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January 1977.

32. Letter, C. Aniel (NRC) to A.E. Scherer (CE), "Evaluation of Topical Report CENPD-138, Supplement 2-P," April 10, 1978.

33. Letter, W.H. Bohlke (FPL) to Document Control Desk (NRC), "St. Lucie Unit 2, Docket No. 50-389, Proposed License Amendment, MTC Change from -27 pcm to -30 pcm," L-91-325, December 17, 1991.

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

34. Letter, J.A. Norris (NRC) to J.H. Goldberg (FPL). "St. Lucie Unit 2 – Issuance of Amendment Re: Moderator Temperature Coefficient (TAC No. M82517)." July 15, 1992.

35. Letter, J.W. Williams, Jr. (FPL) to D.G. Eisenhut (NRC). "St. Lucie Unit No. 2, Docket No. 50-389, Proposed License Amendment, Cycle 2 Reload." L-84-148, June 4, 1984.

36. Letter, J.R. Miller (NRC) to J.W. Williams, Jr. (FPL), Docket No. 50-389, Regarding Unit 2 Cycle 2 License Approval (Amendment No. 8 to NPF-16 and SER), November 9, 1984 (Approval to Methodology contained in L-84-148).

37. Letter, A.E. Scherer Enclosure 1-P to LD-82-001, "CESEC-Digital Simulation of a Combustion Engineering Nuclear Steam Supply System," December 1981.

38. Safety Evaluation Report, "CESEC Digital Simulation of a Combustion Engineering Steam Supply System (TAC No.: 01142)," October 27, 1983.

39. CENPD-282-P-A, Volumes 1, 2 and 3, and Supplement 1, "Technical Manual for the CENTS Code," February 1991, February 1991, October 1991, and June 1993, respectively.

40. CEN-121(B)-P, "CEAW, Method of Analyzing Sequential Control Element Assembly Group Withdrawal Event for Analog Protected Systems," November 1979 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

41. CEN-133(B), "FIESTA, A One Dimensional, Two Group Space-Time Kinetics Code for Calculating PWR Scram Reactivities," November 1979 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

42. CEN-348(B)-P-A, Supplement 1-P-A, "Extended Statistical Combination of Uncertainties," January 1997.

43. CEN-372-P-A, "Fuel Rod Maximum Allowable Gas Pressure," May 1990.

44. CENPD-183-A, "C-E Methods for Loss of Flow Analysis," June 1984.

45. CENPD-190-A, "C-E Method for Control Element Assembly Ejection Analysis," July 1976.

CORE OPERATING LIMITS REPORT (COLR) (continued)

b. (continued)

46. CENPD-199-P, Rev. 1-P-A, Supplement 2-P-A, "CE Setpoint Methodology," June 1998.

47. CENPD-382-P-A, "Methodology for Core Designs Containing Erbium Burnable Absorbers," August 1993.

48. CEN-396(L)-P, "Verification of the Acceptability of a 1-Pin Burnup Limit of 60 MWD/KG for St. Lucie Unit 2," November 1989 (NRC SER dated October 18, 1991, Letter J.A. Norris (NRC) to J.H. Goldberg (FPL), TAC No. 75947).

49. CENPD-269-P, Rev. 1-P, "Extended Burnup Operation of Combustion Engineering PWR Fuel," July 1984.

50. CEN-289(A)-P, "Revised Rod Bow Penalties for Arkansas Nuclear One Unit 2," December 1984 (NRC SER dated December 21, 1999, Letter K. N. Jabbour (NRC) to T.F. Plunkett (FPL), TAC No. MA4523).

51. CENPD-137, Supplement 2-P-A, "Calculative Methods for the ABB CE Small Break LOCA Evaluation Model," April 1998.

52. CENPD-140-A, "Description of the CONTRANS Digital Computer Code for Containment Pressure and Temperature Transient Analysis," June 1976.

53. CEN-365(L), "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis," June 1988 (NRC SER dated March 13, 1989, Letter J.A. Norris (NRC) to W.F. Conway (FPL), TAC No. 69325).

54. DP-456, F.M. Stern (CE) to E. Case (NRC), dated August 19, 1974, Appendix 6B to CESSAR System 80 PSAR (NRC SER, NUREG-75/112, Docket No. STN 50-470, "NRC SER – Standard Reference System, CESSAR System 80," December 1975).

55. CENPD-387-P-A, Revision 000, "ABB Critical Heat Flux Correlations for PWR Fuel," May 2000.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SHUTDOWN MARGIN, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any mid cycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the NRC within the time period specified for each report.

6.10 RECORD RETENTION

In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.1 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment related to nuclear safety.

c. All REPORTABLE EVENTS.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

(continued on page 6-21)

RECORD RETENTION (Continued)

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

6.10.2 The following records shall be retained for the duration of the unit Operating License:

a. Records and drawing changes reflecting unit design modifications made to systems and equipment described in the Final Safety Analysis Report.

b. Records of new and irradiated fuel inventory, fuel transfers, and assembly burnup histories.

c. Records of reactor tests and experiments.

d. Records of radiation exposure for all individuals entering radiation control areas.

e. Records of gaseous and liquid radioactive material released to the environs.

f. Records of transient or operational cycles for those unit components identified in Table 5.7-1.

g. Records of training and qualification for current members of the unit staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of quality assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the FRG and the CNRB.

l. Records of the service lives of all snubbers including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

n. Annual Radiological Environmental Operating Reports; and records of analyses transmitted to the licensee which are used to prepare the Annual Radiological Environmental Monitoring Report.

o. Meteorological data, summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23.

p. Records of audits performed under the requirements of Specifications 6.5.2.8 and 6.8.4.

q. Records of reviews performed for changes made to the OFFSITE DOSE CALCULATION MANUAL and the PROCESS CONTROL PROGRAM.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.1601(a) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr measured at a distance of 30 cm (12 in) shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. Any individual or group of individuals permitted to enter such areas shall be provided with or accompanied by one or more of the following:

a. A radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A radiation monitoring device which continuously integrates the radiation dose rate in the area and alarms when a preset integrated dose is received. Entry into such areas with this monitoring device may be made after the dose rate level in the area has been established and personnel have been made knowledgeable of them.

c. A health physics qualified individual (i.e., qualified in radiation protection procedures) with a radiation dose rate monitoring device who is responsible for providing positive control over the activities within the area and shall perform periodic radiation surveillance at the frequency specified by the facility Health Physicist in the RWP.

6.12.2 In addition to the requirements of Specification 6.12.1, areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter shall be provided with locked doors to prevent unauthorized entry, and the keys shall be maintained under the administrative control of the Shift Foreman on duty and/or health physics supervision. Doors shall remain locked except during periods of access by personnel under an approved RWP which shall specify the dose rate levels in the immediate work area and the maximum allowable stay time for individuals in that area. For individual areas accessible to personnel with radiation levels in excess of 1000 mrem/hr at 30 cm (12 in) and less than 500 rads/hr at 1 meter that are located within large areas, such as PWR containment, where no enclosure exists for purposes of locking, and no enclosure can be reasonably constructed around the individual areas, then that area shall be roped off, conspicuously posted and a flashing light shall be activated as a warning device. In lieu of the stay time specification of the RWP, direct or remote (such as use of closed circuit TV cameras) continuous surveillance may be made by personnel qualified in radiation protection procedures to provide positive exposure control over the activities within the area.

*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 are otherwise following plant radiation protection procedures for entry into high radiation areas.

6.13 PROCESS CONTROL PROGRAM (PCP)

Changes to the PCP:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2q. This documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b) A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

2. Shall become effective after review and acceptance by the Facility Review Group and the approval of the Plant General Manager.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

Changes to the OCDM:

1. Shall be documented and records of reviews performed shall be retained as required by Specification 6.10.2q. This documentation shall contain:

   a) Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and

   b) A determination that the change will maintain the level of radioactive effluent control required by 10 CFR 20.1302, 40 CFR Part 190, 10 CFR 50.36a, and Appendix I to 10 CFR Part 50 and not adversely impact the accuracy or reliability of effluent, dose, or setpoint calculations.

2. Shall become effective after review and acceptance by the Facility Review Group and the approval of the Plant General Manager.

3. Shall be submitted to the Commission in the form of a complete, legible copy of the entire ODCM as a part of or concurrent with the Annual Radioactive Effluent Release Report for the period of the report in which any change to the ODCM was made. Each change shall be identified by markings in the margin of the affected pages, clearly indicating the area of the page that was changed, and shall indicate the date (e.g., month/year) the change was implemented.