NUREG-0949

Technical Specifications
St. Lucie Plant
Unit No. 2

Docket No. 50-389

Appendix "A" to
License No. NPF-16

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

Office of Nuclear Reactor Regulation

April 1983

UNITED STATES NUCLEAR REGULATORY COMMISSION

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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.

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

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 as provided in Table 3.6-2 of Specification 3.6.3.

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

DOSE EQUIVALENT I-131

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

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

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

ENGINEERED SAFETY 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.

1.16 LOW TEMPERATURE RCS OVERPRESSURE PROTECTION RANGE

The LOW TEMPERATURE RCS OVERPRESSURE PROTECTIVE RANGE is that operating condition when (1) the cold leg temperature is $\leq$ 280°F during cooldown and $\leq$ 320°F during heatup and (2) the Reactor Coolant System has pressure boundary integrity. The Reactor Coolant System does not have pressure boundary integrity when the Reactor Coolant System is open to containment and the minimum area of the Reactor Coolant System opening is greater than 3.58 square inches.

MEMBER(S) OF THE PUBLIC

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

OFFSITE DOSE CALCULATION MANUAL (ODCM)

1.18 The OFFSITE DOSE CALCULATION MANUAL shall contain the current methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring alarm/trip setpoints, and shall include the Radiological Environmental Monitoring Sample point locations.

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 shall contain the provisions, based on full scale testing, to assure that dewatering of spent bead resins results in a waste form with the properties that meet the requirements of 10 CFR Part 61 (as implemented by 10 CFR Part 20) and of the low level radioactive waste disposal site at the time of disposal.

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 2560 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 OCCURRENCE

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

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 The SITE BOUNDARY shall be that line beyond which the land is neither owned, leased, nor 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 An UNRESTRICTED AREA shall be any area at or beyond the SITE BOUNDARY access to which is not controlled by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials, or any area within the SITE BOUNDARY used for residential quarters or for industrial, commercial, institutional, and/or recreational purposes.

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.

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.

PEAK LINEAR HEAT RATE

2.1.1.2 The peak linear heat rate of the fuel shall be maintained less than or equal to 21.0 kW/ft (value corresponding to centerline fuel melt).

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the peak linear heat rate of the fuel has exceeded 21.0 kW/ft (value corresponding to centerline fuel melt), 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.

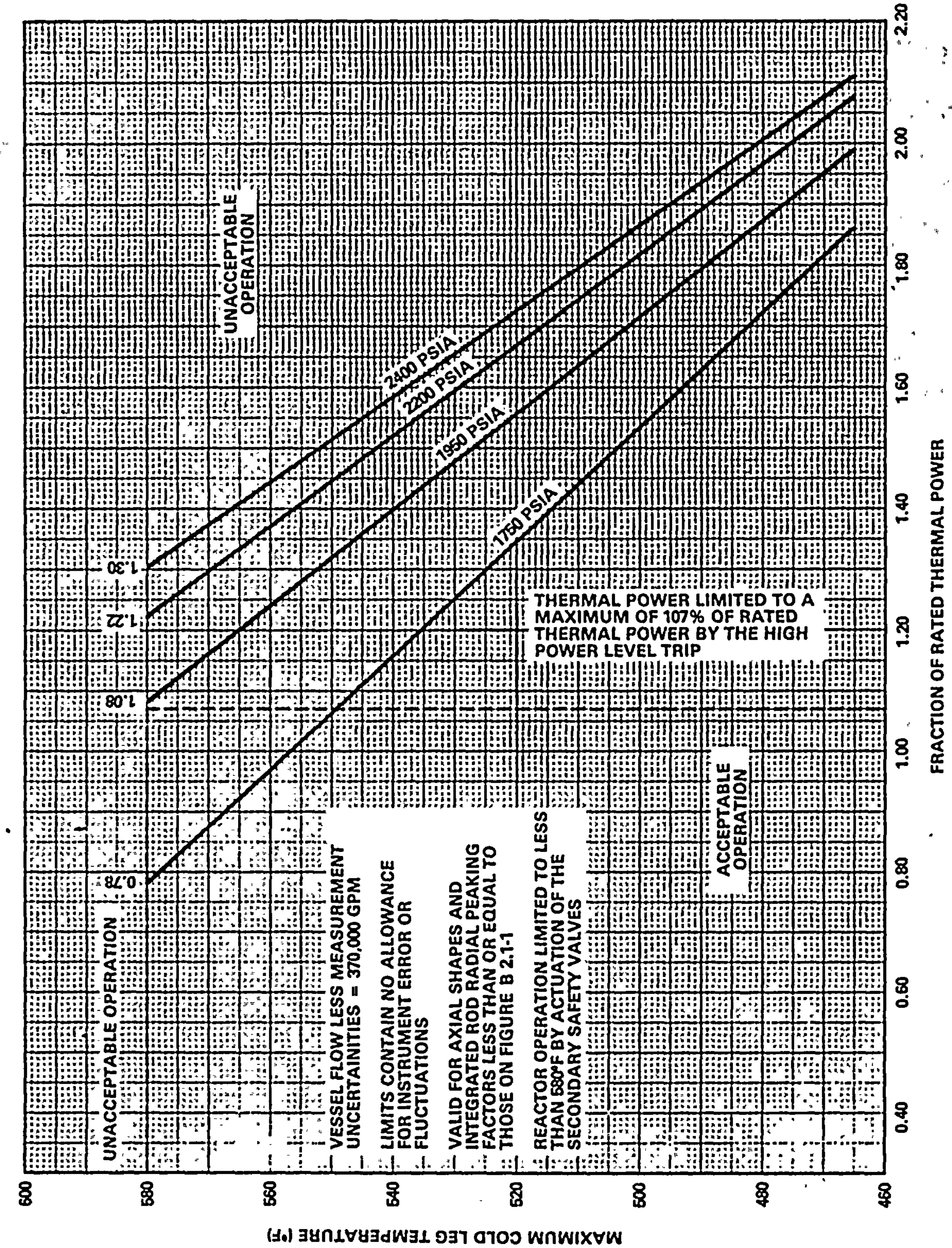

Figure 2.1-1
Reactor core thermal margin safety limit lines
Four reactor coolant pumps operating

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	$\leq$ 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.	$\leq$ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.
3. Pressurizer Pressure - High	$\leq$ 2370 psia	$\leq$ 2374 psia
4. Thermal Margin/Low Pressure		
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	$\leq$ 4.0 psig	$\leq$ 5.0 psig
6. Steam Generator Pressure - Low	$\geq$ 626.0 psia (2)	$\geq$ 621.0 psia (2)
7. Steam Generator Pressure(1) Difference - High (Logic in TM/LP Trip Unit)	$\leq$ 120.0 psid	$\leq$ 132.0 psid
8. Steam Generator Level - Low	$\geq$ 39.5% (3)	$\geq$ 39.1% (3)

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Local Power Density - High(5)	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	$\geq$ 636 gpm**	$\geq$ 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)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute
14. Reactor Coolant Flow - Low	$\geq$ 95.4% of design Reactor Coolant flow with four pumps operating*	$\geq$ 94.9% of design Reactor Coolant flow with four pumps operating*
15. Loss of Load (Turbine) Hydraulic Fluid Pressure - Low(5)	$\geq$ 800 psig	$\geq$ 800 psig

*Design reactor coolant flow with four pumps operating is 370,000 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 THERMAL 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 THERMAL POWER is $\geq 10^{-4}$% or $\leq$ 15% of RATED THERMAL POWER.

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

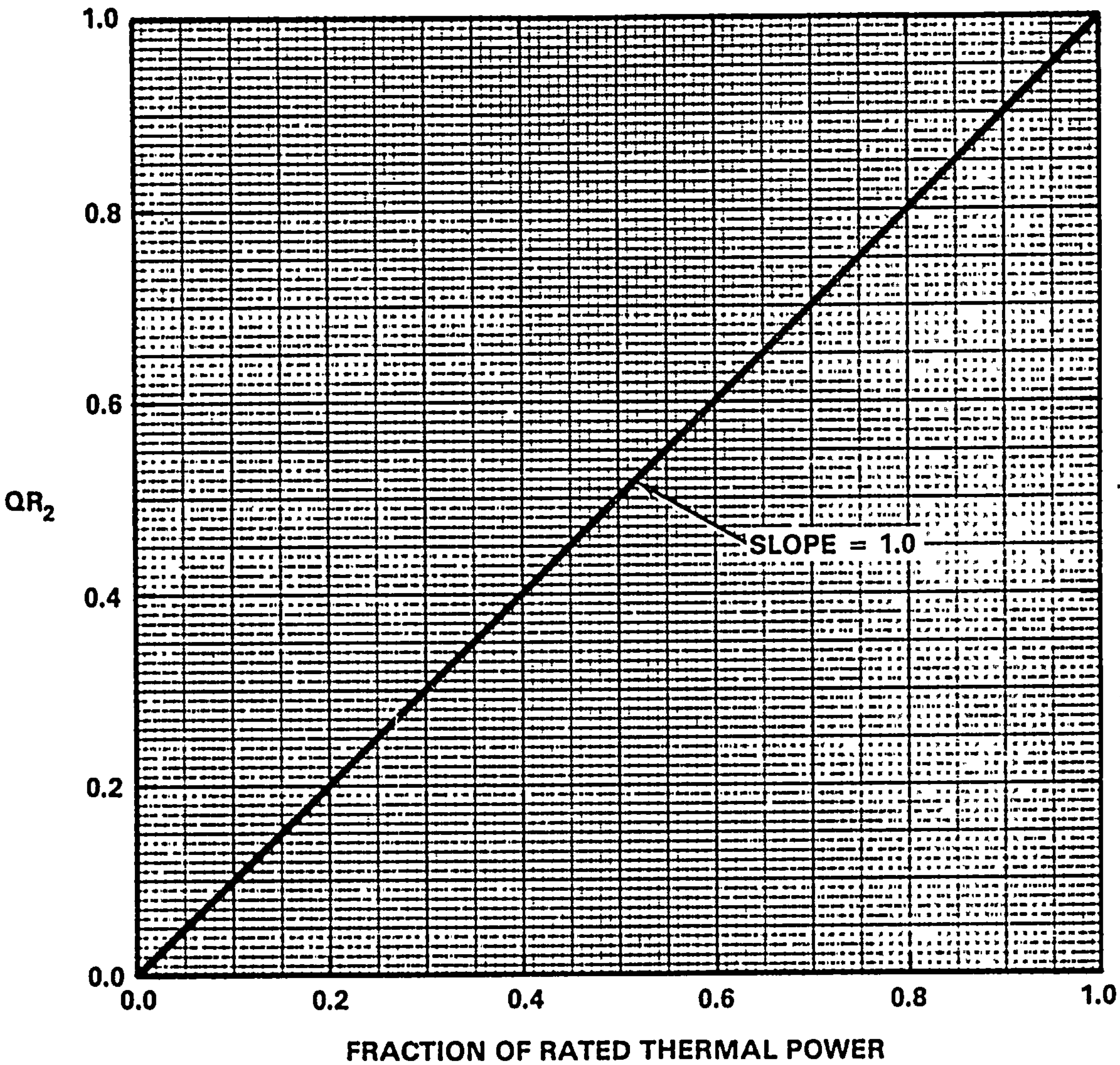

Figure 2.2-1
Local power density - High trip setpoint
Part 1 (Fraction of RATED THERMAL POWER versus QR_2)

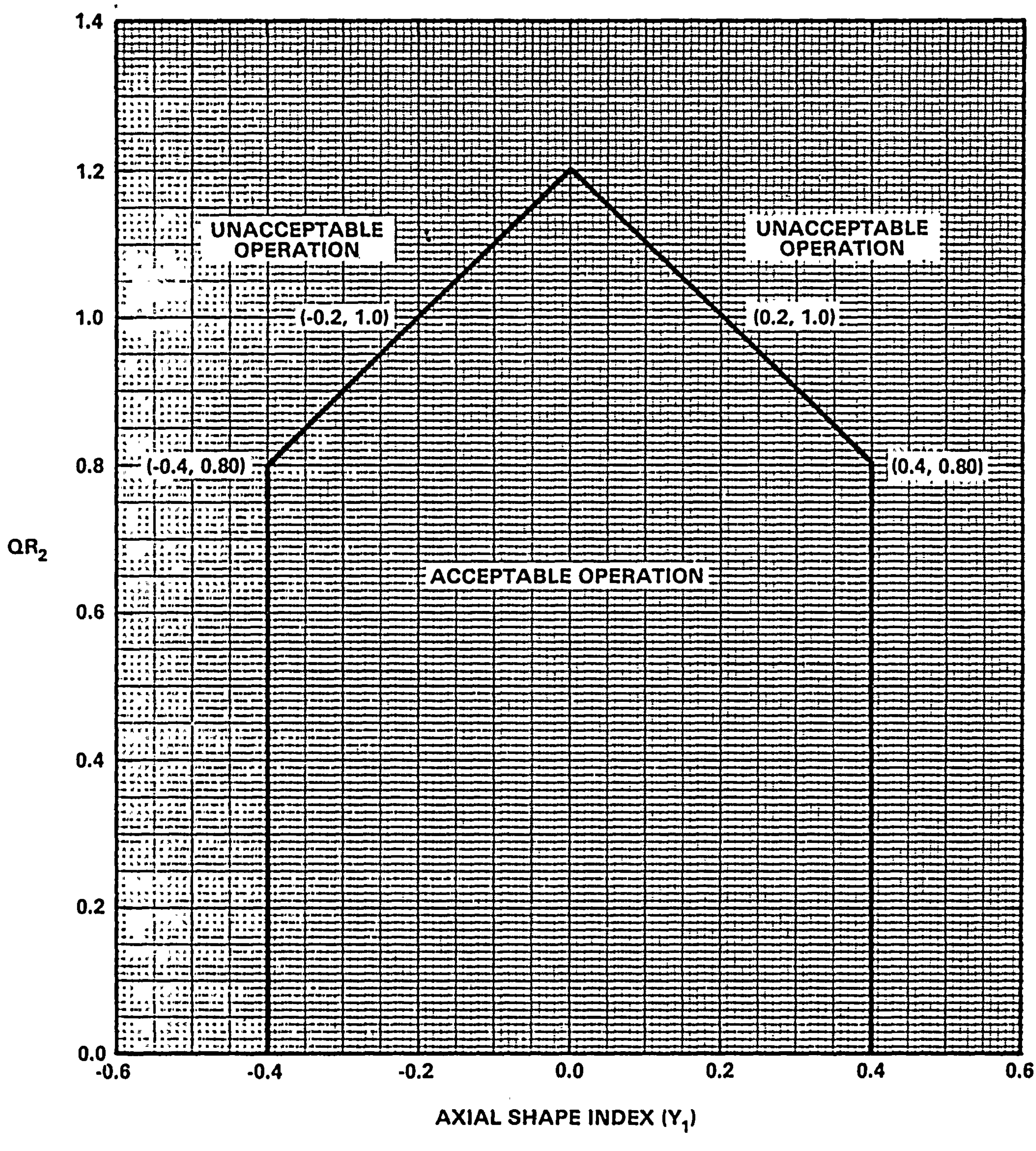

Figure 2.2-2
Local power density-high trip setpoint
Part 2 (QR_2 versus Y_1)

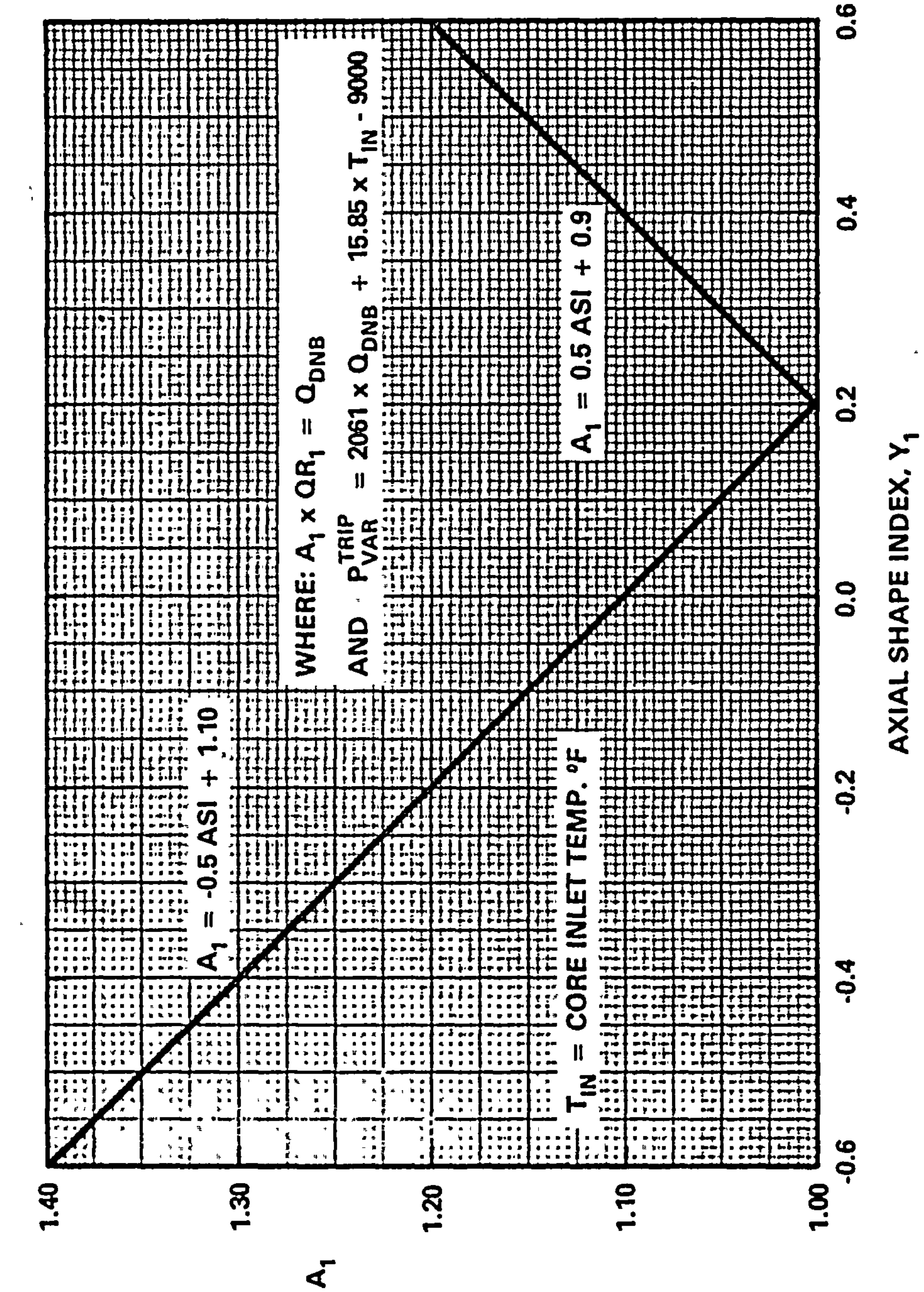

Figure 2.2-3
Thermal margin/low pressure trip setpoint
Part 1 (Y_1 versus A_1)

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

AND $P_{VAR}^{TRIP} = 2061 \times Q_{DNB} + 15.85 \times T_{IN} - 9000$

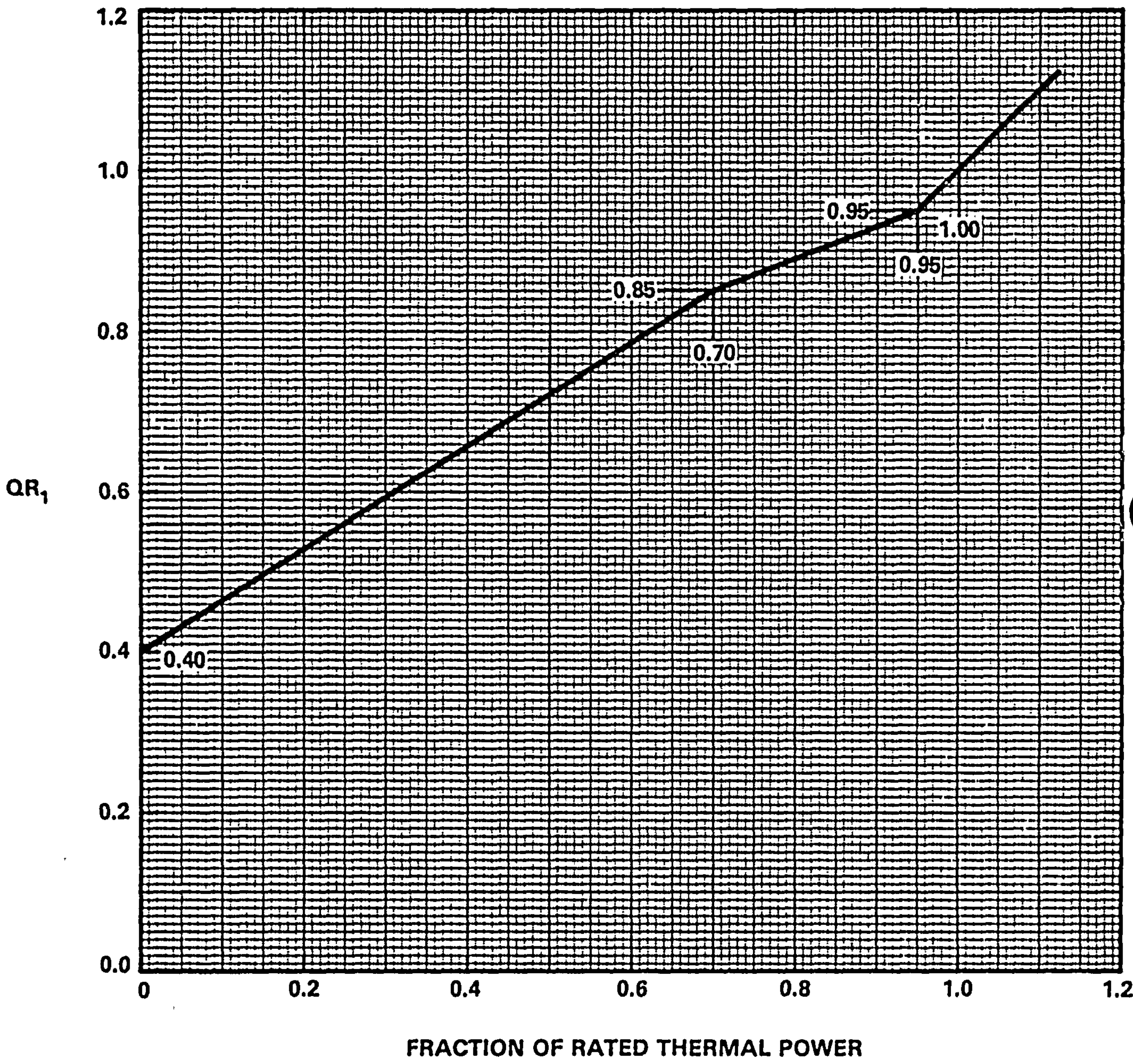

Figure 2.2-4
Thermal margin/low pressure trip setpoint
Part 2 (Fraction of RATED THERMAL POWER versus QR_1)

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

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

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady-state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the CE-1 correlation. The CE-1 DNB correlation has been developed to predict the DNB heat flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.20. This value corresponds to a 95% probability at a 95% confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than 1.20 for the family of axial shapes and corresponding radial peaks shown in Figure B 2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.2-1. The area of safe operation is below and to the left of these lines.

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The Thermal Margin/Low Pressure and Local Power Density Trip Systems, in conjunction with Limiting Conditions for Operation, the Variable Overpower Trip and the Power Dependent Insertion Limits, assure that the Specified Acceptable Fuel Design Limits on DNB and Fuel Centerline Melt are not exceeded during normal operation and design basis Anticipated Operation Occurrences.

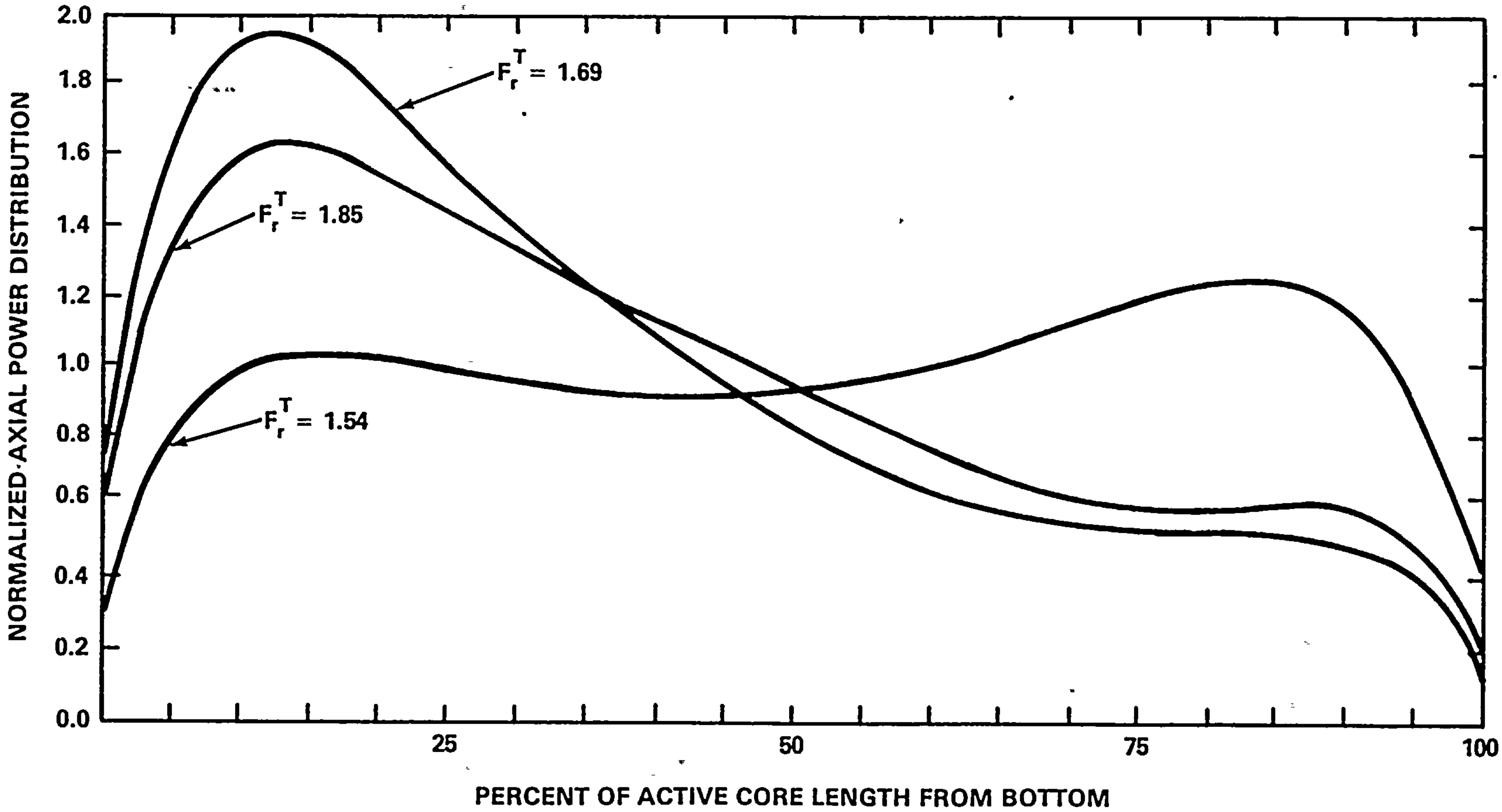

Figure B 2.1-1
Axial power distribution for thermal margin safety limits

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The Reactor Coolant System components are designed to Section III, 1971 Edition including Addenda to the Summer, 1973, of the ASME Code for Nuclear Power Plant Components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each functional unit. The Trip Setpoints have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their Safety Limits during normal operation and design basis anticipated operational occurrences and to assist the Engineered Safety Features Actuation System in mitigating the consequences of accidents. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that the difference between each Trip Setpoint and the Allowable Value is equal to or less than the drift allowance assumed for each trip in the safety analyses.

Manual Reactor Trip

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

Variable Power Level-High

A Reactor trip on Variable Overpower is provided to protect the reactor core during rapid positive reactivity addition excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure Trip.

The Variable Power Level High trip setpoint is operator adjustable and can be set no higher than 9.61% above the indicated THERMAL POWER level. Operator action is required to increase the trip setpoint as THERMAL POWER is increased. The trip setpoint is automatically decreased as THERMAL POWER decreases. The trip setpoint has a maximum value of 107.0% of RATED THERMAL POWER and a minimum setpoint of 15.0% of RATED THERMAL POWER. Adding to this maximum value the possible variation in trip point due to calibration and instrument errors, the maximum actual steady-state THERMAL POWER level at which a trip would be actuated is 112% of RATED THERMAL POWER, which is the value used in the safety analyses.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, in conjunction with the pressurizer safety valves and main steam safety valves, provides Reactor Coolant System protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is at less than or equal to 2375 psia which is below the nominal lift setting 2500 psia of the pressurizer safety valves and its operation minimizes the undesirable operation of the pressurizer safety valves.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.20.

The trip is initiated whenever the Reactor Coolant System pressure signal drops below either 1900 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

The Thermal Margin/Low Pressure trip setpoints are derived from the core safety limits through application of appropriate allowances for equipment response time measurement uncertainties and processing error. A safety margin is provided which includes: an allowance of 2.0% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 3.0°F to compensate for potential temperature measurement uncertainty; and a further allowance of 91.0 psia to compensate for pressure measurement error and time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the safety limit. The 91.0 psia allowance is made up of a 25.0 psia pressure measurement allowance and a 66.0 psia time delay allowance.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip is initiated prior to or concurrently with a safety injection (SIAS). This also provides assurance that a reactor trip is initiated prior to or concurrently with an MSIS.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setpoint of 620 psia is sufficiently below the full load operating point of approximately 885 psia so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of 30 psi in the safety analyses.

Steam Generator Level-Low

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

Local Power Density-High

The Local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower excore neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15% power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

RCP Loss of Component Cooling Water

A loss of component cooling water to the reactor coolant pumps causes a delayed reactor trip. This trip provides protection to the reactor coolant pumps by ensuring that plant operation is not continued without cooling water available. The trip is delayed 10 minutes following a reduction in flow to below the trip setpoint and the trip does not occur if flow is restored before 10 minutes elapses. No credit was taken for this trip in the safety analysis. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protective System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Reactor Coolant Flow - Low

The Reactor Coolant Flow - Low trip provides protection against a reactor coolant pump sheared shaft event and a two pump opposite loop flow coastdown event. A trip is initiated when the pressure differential across the primary side of either steam generator decreases below a variable setpoint. This variable setpoint stays a set amount below the pressure differential unless limited by a set maximum decrease rate or a set minimum value. The specified setpoint ensures that a reactor trip occurs to prevent violation of local power density or DNBR safety limits under the stated conditions.

Loss of Load (Turbine)

The Loss of Load (Turbine) trip is provided to trip the reactor when the turbine is tripped above a predetermined power level. This trip is an equipment protective trip only and is not required for plant safety. This trip's setpoint does not correspond to a Safety Limit and no credit was taken in the safety analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Asymmetric Steam Generator Transient Protection Trip Function (ASGTPTF)

The ASGTPTF utilizes steam generator pressure inputs to the TM/LP calculator, which causes a reactor trip when the difference in pressure between the two steam generators exceeds the trip setpoint. The ASGTPTF is designed to provide a reactor trip for those Anticipated Operational Occurrences associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator caused by a single Main Steam Isolation Valve closure.

The equipment trip setpoint and allowable values are calculated to account for instrument uncertainties, and will ensure a trip at or before reaching the analysis setpoint.

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 unless the conditions of the Limiting Condition for Operation are met without reliance on provisions contained in the ACTION requirements. This provision shall not prevent passage through or to OPERATIONAL MODES as required to comply with ACTION 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.

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

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

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

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

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

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

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

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

APPLICABILITY

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

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

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

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

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

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

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

With the SHUTDOWN MARGIN less than 5.0% delta k/k, immediately initiate and continue boration at greater than or equal to 40 gpm of a solution containing greater than or equal to 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be greater than or equal to 5.0% delta k/k:

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

b. When in MODE 1 or MODE 2 with K_{eff} greater than or equal to 1.0, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

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

*See Special Test Exception 3.10.1.

SURVEILLANCE REQUIREMENTS (Continued)

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e. below, with the CEA groups at the 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 ± 1.0% delta k/k at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1e., 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.

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

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Within 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:

a. Less positive than $+0.5 \times 10^{-4}$ delta k/k/°F at $\leq$ 70% RATED THERMAL POWER,

b. Less positive than 0.0×10^{-4} delta k/k/°F at > 70% RATED THERMAL POWER, and

c. Less negative than -2.7×10^{-4} delta k/k/°F at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 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.

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths and one associated heat tracing circuit shall be OPERABLE 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 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 7 days by verifying that the temperature of the heat traced portion of the flow path is above the temperature limit line shown on Figure 3.1-1 when a flow path from the boric acid makeup tanks is used.

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one of the above required boron injection flow paths to the Reactor Coolant System OPERABLE, restore at least two boron injection flow paths to the Reactor Coolant System to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2.0% delta k/k at 200°F within the next 6 hours; restore at least two flow paths to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

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

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

CHARGING PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one charging pump or one high pressure safety injection pump in the boron injection flow path required OPERABLE pursuant to Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency 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 pump 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 Specification 4.0.5.

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two charging pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to at least 2.0% delta k/k at 200°F within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4.1 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 Specification 4.0.5.

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

REACTIVITY CONTROL SYSTEMS

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 makeup 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 Specification 4.0.5.

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 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump(s) in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid makeup pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid makeup pump to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2.0% delta k/k at 200°F; restore the above required boric acid makeup pump(s) to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.6 The above required boric acid makeup pump(s) shall be demonstrated OPERABLE by verifying, that on recirculation flow, the pump(s) develop a discharge pressure of greater than or equal to 90 psig when tested pursuant to Specification 4.0.5.

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 and at least one associated heat tracing circuit with a minimum contained volume of 4150 gallons of 8 weight percent 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

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

b. At least once per 24 hours by verifying the RWT temperature when it is the source of borated water and the outside air temperature is outside the range of 40°F and 120°F.

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

b. The refueling water 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 between 55°F and 100°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the above required boric acid makeup tank inoperable, restore the tank to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and borated to a SHUTDOWN MARGIN equivalent to at least 2.0% delta k/k at 200°F; restore the above required boric acid makeup tank to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With the refueling water tank inoperable, restore the tank to OPERABLE status within 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 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

   1. Verifying the boron concentration in the water,
   2. Verifying the contained borated water volume of the water source, and
   3. Verifying the boric acid makeup tank solution temperature.

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.

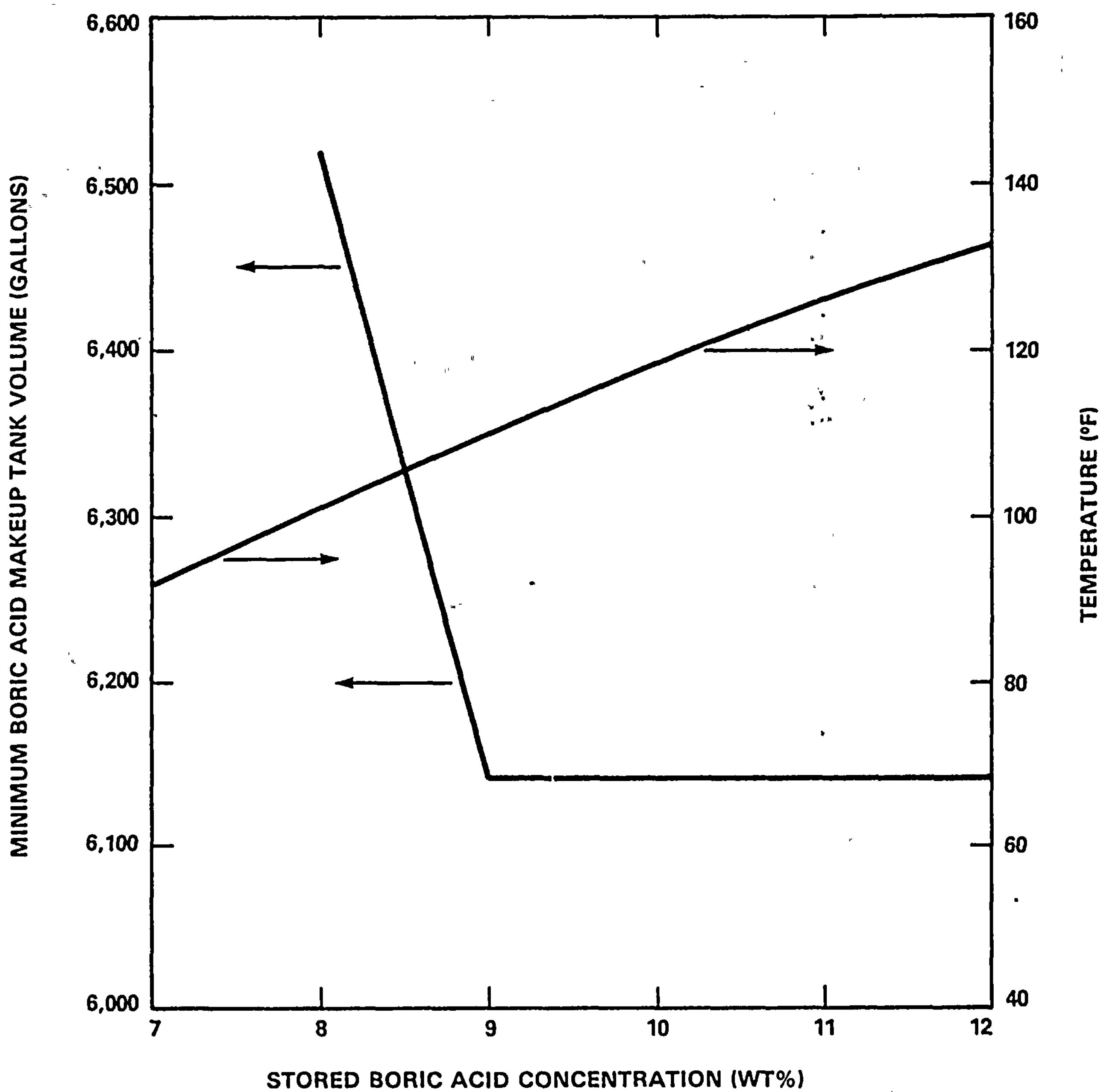

Figure 3.1-1
.Minimum boric acid makeup tank volume and temperature as a function of stored boric acid concentration

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.2.9 Boron concentration shall be verified consistent with SHUTDOWN MARGIN requirements of Specifications 3.1.1.1, 3.1.1.2, and 3.9.1.

APPLICABILITY:

a. MODES 3, 4, and 5 with RCS level above the hot leg centerline by use of boronometer or sampling per Table 3.1-1, and

b. MODE 5 with RCS level below the hot leg centerline; and MODE 6 by sampling per Table 3.1-1.

ACTION:

a. With the boron concentration not consistent with required SHUTDOWN MARGIN, initiate emergency boration.

b. If unable to determine the RCS boron concentration by the means specified above, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes until one of the means of determining the RCS boron concentration as specified above is restored to OPERABLE status.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.1.2.9

a. When in MODES 3, 4, 5, and 6, the boron concentration shall be determined consistent with SHUTDOWN MARGIN requirements once per 8 hours.

b. The boronometer, when used to monitor boron concentration, shall be demonstrated OPERABLE by performance of:

   1. a CHANNEL CHECK once per 12 hours, and

   2. a CHANNEL CALIBRATION once per 18 months.

c. Whenever performing an RCS heatup or cooldown, determine the boron concentration at least once every 50°F change in temperature.

TABLE 3.1-1

MONITORING FREQUENCIES FOR BACKUP BORON DILUTION DETECTION FOR ST. LUCIE-2

MODE	Number of OPERABLE Charging Pumps* 0	1	2	3
3	12 hr	4 hr	2 hr	1 hr
4	12 hr	4 hr	2 hr	1 hr
5	8 hr	1 hr	0.5 hr	0.5 hr
5 (RCS level below hot leg centerline)	8 hr	0.5 hr	Operation not allowed**	Operation not allowed**
6	24 hr	4 hr	2 hr	1 hr

*Charging pump OPERABILITY for any period of time shall constitute OPERABILITY for the entire monitoring frequency.

**In MODE 5 with the RCS level below the hot leg centerline, at least two charging pumps shall be verified to be inoperable by racking out their motor circuit breakers.

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, 4, and 5 fully withdrawn and the CEA's in group 6 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 within 1# hour the misaligned CEA is either:

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

---

*See Special Test Exceptions 3.10.2, 3.10.4, and 3.10.5.

#With the CEA not positioned within 15 inches of all the CEAs in its group within (a) 15 minutes with previous $F_r^T > 1.5$, or, (b) 25 minutes with previous $F_r^T \le 1.5$, reduce power to $\le$ 70% of RATED THERMAL POWER prior to completing ACTION d.1. and d.2.

ACTION: (Continued)

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 Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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

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

e. With one 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 Figure 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

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

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

f. 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.

g. With one full-length CEA inoperable due to causes other than addressed by ACTION a., above, but within its above specified alignment requirements and either fully withdrawn or within the Long Term Steady State Insertion Limits if in full-length CEA group 6, operation in MODES 1 and 2 may continue.

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

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 31 days.

4.1.3.1.3 The CEA Block Circuit shall be demonstrated OPERABLE at least once per 31 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 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 31 days, and

b. At least once per 6 months.

---

*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 3.0 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.

*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 Figure 3.1-2 (regulating CEAs are considered to be fully withdrawn in accordance with 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 using the above figure.

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 of Figure 3.1-2 are not exceeded, or

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

---

*See Special Test Exceptions 3.10.2, 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.

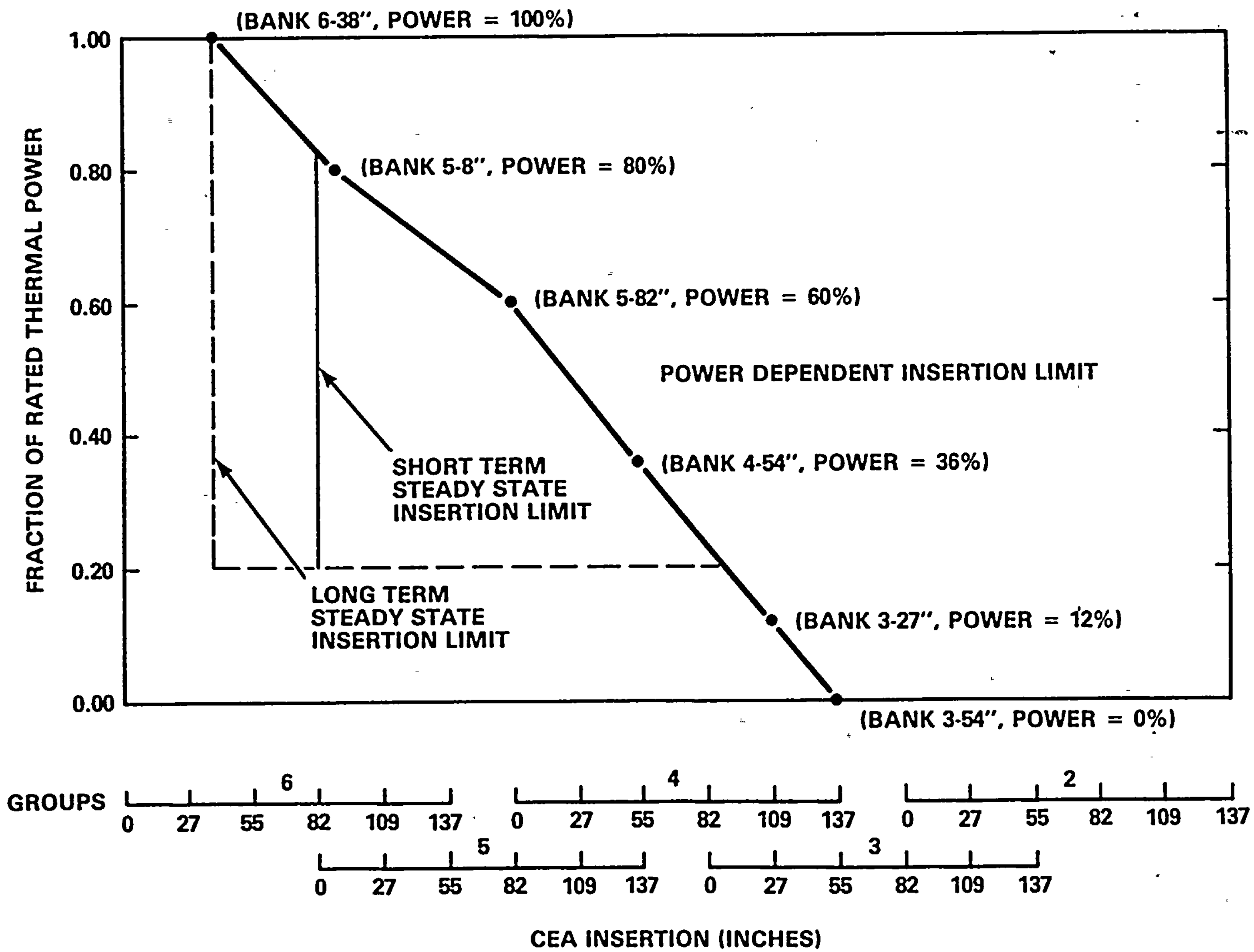

Figure 3.1-2
CEA insertion limits vs THERMAL POWER with four reactor coolant pumps operating

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 shown on Figure 3.2-1.

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 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 core power distribution 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 Figure 3.2-2.

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

$M \times N$

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_{xy}^{T} curve of Figure 3.2-3.

4.2.1.4 Incore Detector Monitoring System# - The incore detector monitoring system may be used for monitoring the core power distribution 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 Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

1. Flux peaking augmentation factors as shown in Figure 4.2-1,

2. A measurement-calculational uncertainty factor of 1.062,

3. An engineering uncertainty factor of 1.03,

4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

5. A THERMAL POWER measurement uncertainty factor of 1.02.

#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.

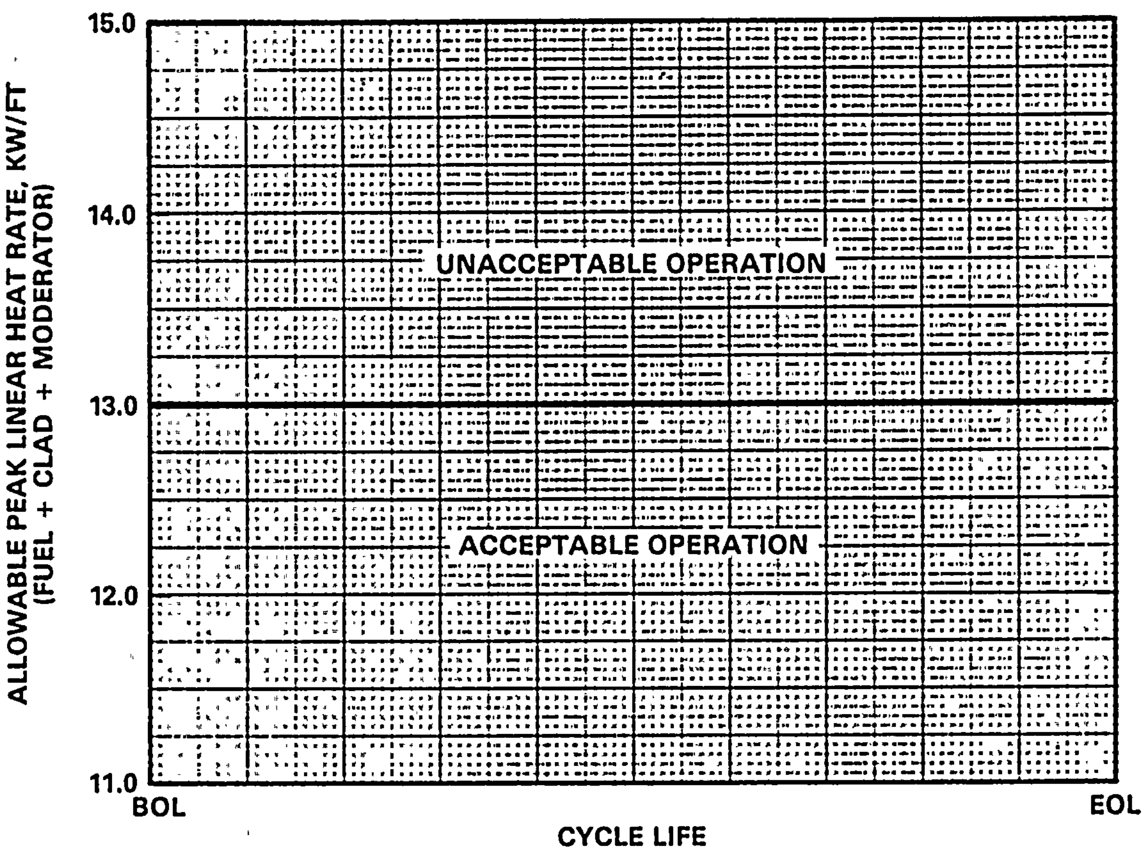

Figure 3.2-1
Allowable peak linear heat rate vs burnup

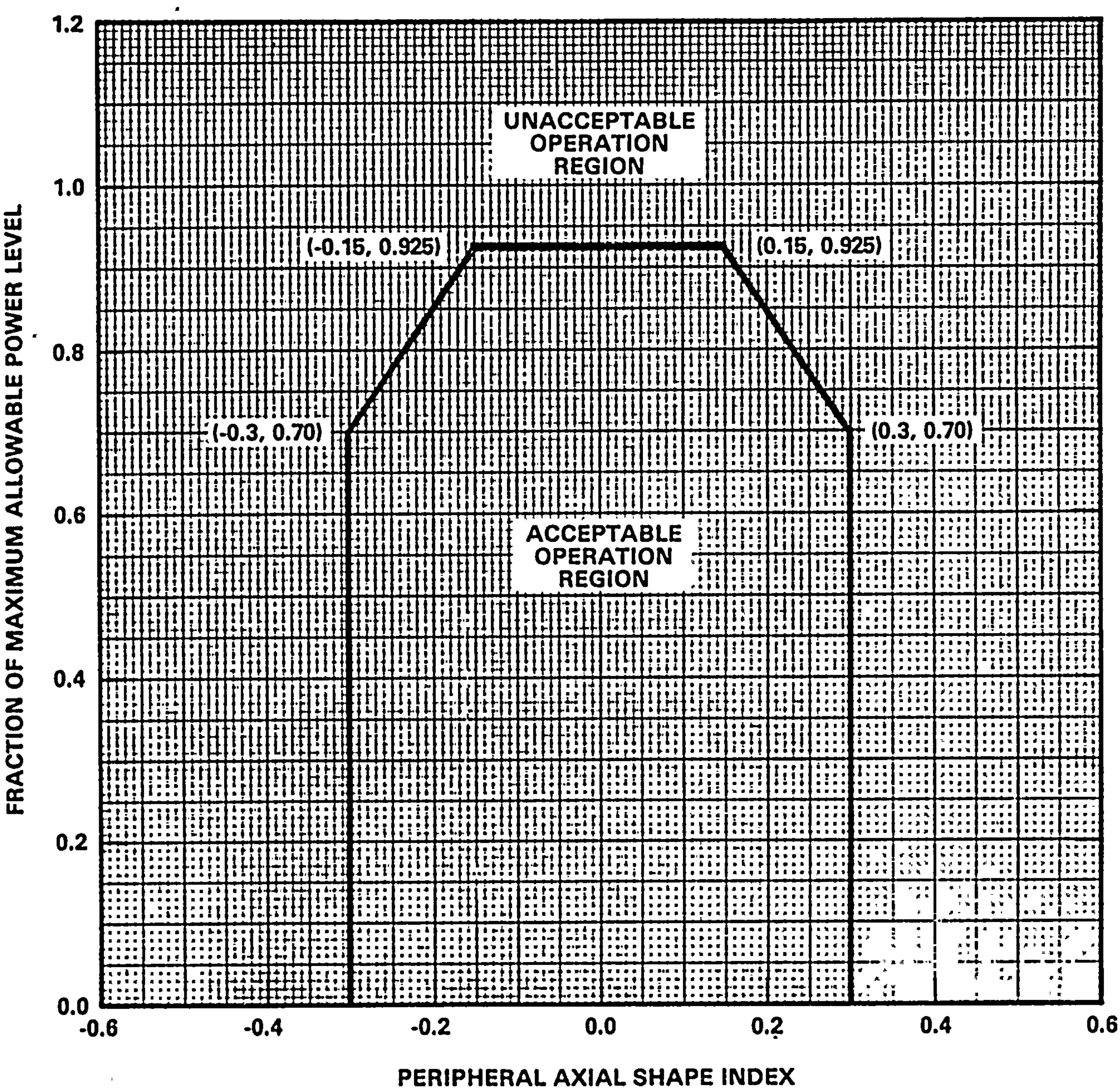

Figure 3.2-2
AXIAL SHAPE INDEX vs fraction of maximum allowable power level per Specification 4.2.1.3

ST. LUCIE - UNIT 2

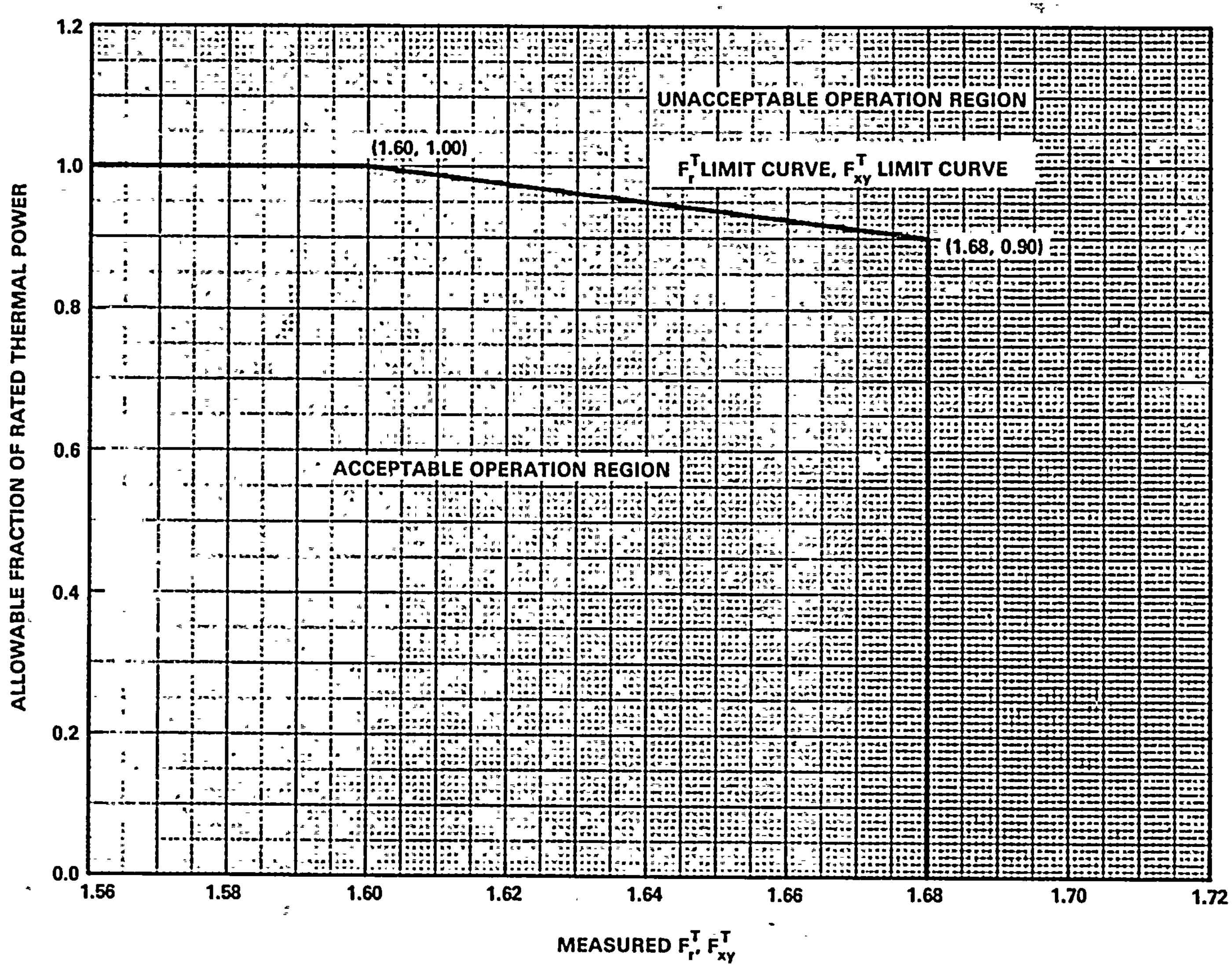

Figure 3.2-3
Allowable combinations of thermal power and F_r^T, F_{xy}^T

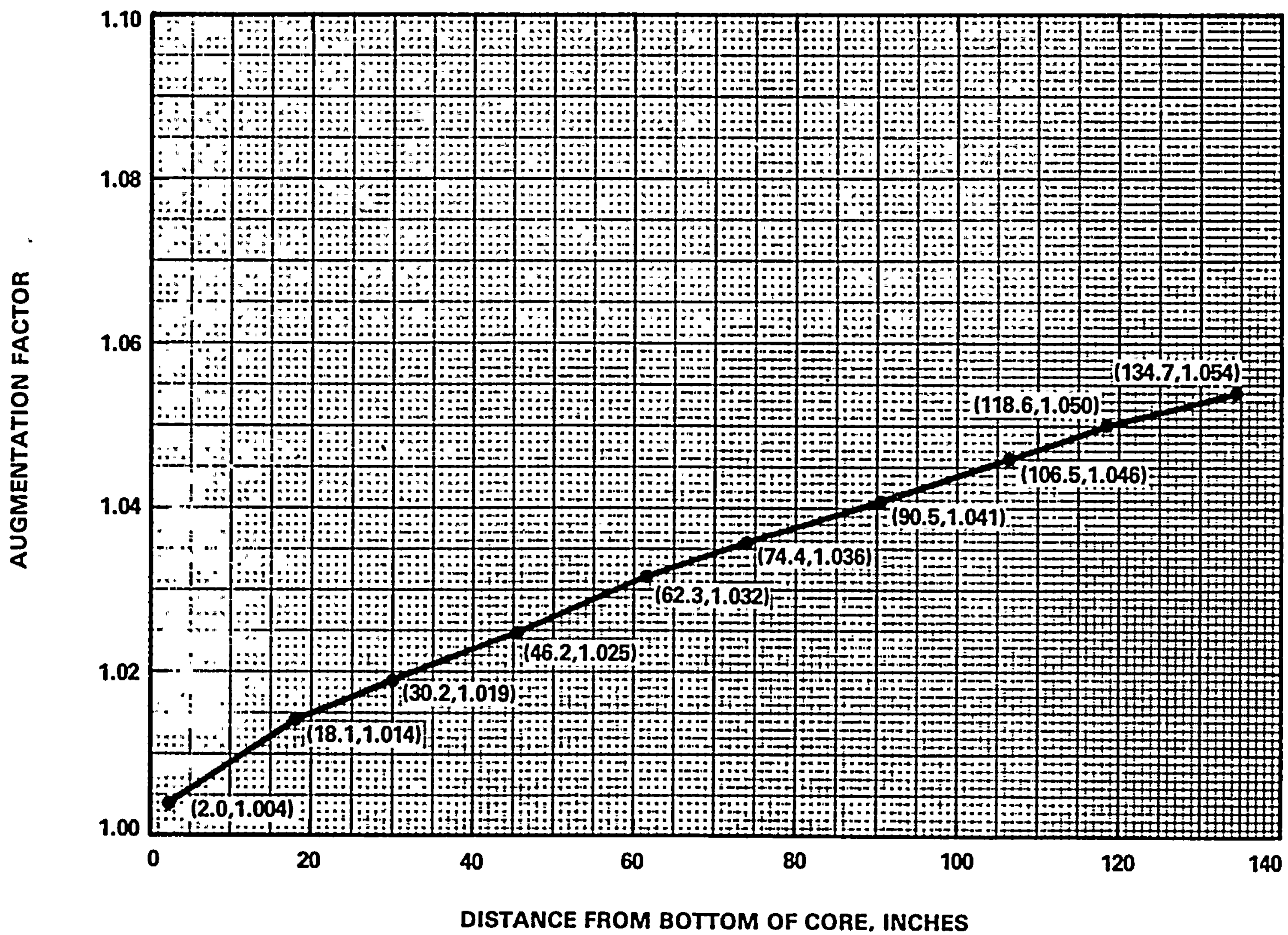

Figure 4.2-1
Augmentation factors vs distance from bottom of core

POWER DISTRIBUTION LIMITS

3/4.2.2 TOTAL PLANAR RADIAL PEAKING FACTORS - F_{xy}^{T}

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F_{xy}^{T} shall be limited to ≤ 1.60.

APPLICABILITY: MODE 1*.

ACTION:

With $F_{xy}^{T} > 1.60$, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_{xy}^{T} to within the limits of 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_{xy}^{T} shall be calculated by the expression $F_{xy}^{T} = F_{xy}(1+T_q)$ when F_{xy} is calculated with a non-full core power distribution analysis code and shall be calculated as $F_{xy}^{T} = F_{xy}$ when calculations are performed with a full core power distribution analysis code. F_{xy}^{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)

4.2.2.3 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 FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, shall be limited to $\leq$ 1.60.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.60, 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 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 Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of 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 $F_r^T = F_r(1+T_q)$ (Rod Bow Penalty)** when F_r is calculated with a non-full core power distribution analysis code and shall be calculated as $F_r^T = F_r$ (Rod Bow Penalty)** 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.

**Rod Bow Penalty in accordance with the requirements of Table 3.2-1.

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.

TABLE 3.2-1

PENALTY TO BE APPLIED TO F_r^T TO ACCOUNT FOR ROD BOW EFFECTS ON DNBR

BURNUP OF BUNDLE (GWd/MTU)	DNBR PENALTY (%)	DNBR PENALTY WITH GRID SPACING PENALTY (%)	PENALTY MULTIPLIER TO BE APPLIED TO MEASURED F_r^T
0-10.0	0.5	1.5	1.013
10.0-20.0	1.0	2.0	1.017
20.0-30.0	2.0	3.0	1.026
30.0-40.0	3.5	4.5	1.038
40.0-50.0	5.5	6.5	1.055

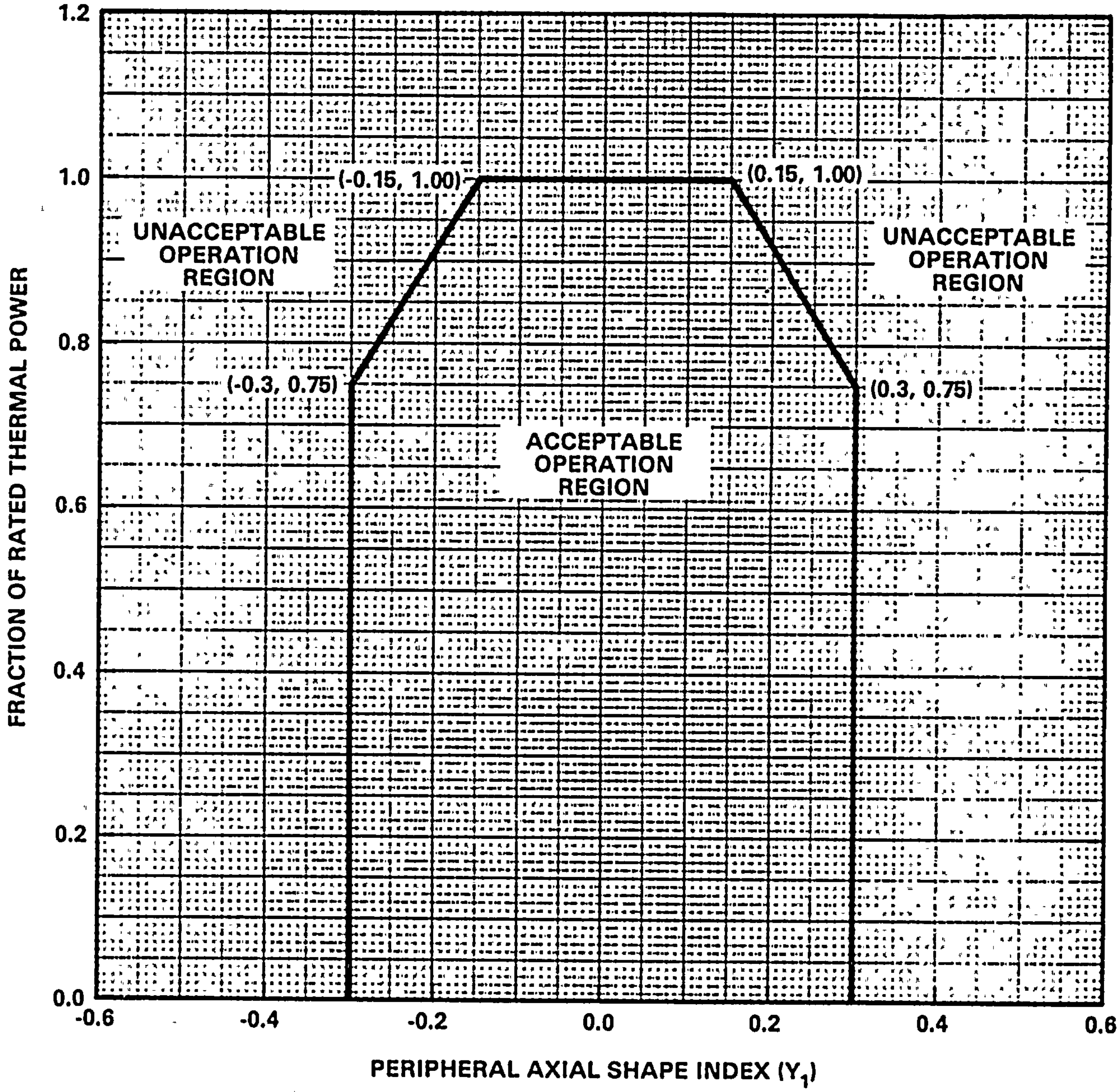

Figure 3.2-4
AXIAL SHAPE INDEX operating limits with four reactor coolant pumps operating

POWER DISTRIBUTION LIMITS

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 $\leq$ 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.

TABLE 3.2-2

DNB MARGIN

	LIMITS
PARAMETER	FOUR REACTOR COOLANT PUMPS OPERATING
Cold Leg Temperature (Narrow Range)	535°F* $\leq$ T $\leq$ 548°F
Pressurizer Pressure	2225 psia** $\leq P_{PZR} \leq$ 2350 psia*
Reactor Coolant Flow Rate	$\geq$ 370,000 gpm
AXIAL SHAPE INDEX	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 with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protective instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations for the MODES and at the frequencies shown in Table 4.3-1.

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

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

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(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(d)	3/SG	1, 2	2#
15. Loss of Load (Turbine Hydraulic Fluid Pressure - Low)	4	2(c)	3	1	2#

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be manually bypassed below 0.5% of RATED THERMAL POWER in conjunction with (d) below; bypass shall be automatically removed when THERMAL 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 THERMAL 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 THERMAL power is $\geq 10^{-4}$% or $\leq$ 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.

TABLE 3.3-1 (Continued)

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	Variable Power Level - High (RPS) Local Power Density - High (RPS) Thermal Margin/Low Pressure (RPS) Rate of Change of Power - High (RPS)
2. Pressurizer Pressure - High	Pressurizer Pressure - High (RPS) Thermal Margin/Low Pressure (RPS)
3. Containment Pressure - High	Containment Pressure - High (RPS) Containment Pressure - High (ESF)
4. Steam Generator Pressure - Low	Steam Generator Pressure - Low (RPS) Steam Generator ΔP 1 and 2 (AFAS 1 and 2) Thermal Margin/Low Pressure (RPS)
5. Steam Generator Level	Steam Generator Level - Low (RPS) Steam Generator ΔP (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 requirement, 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.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Variable Power Level - High	$\leq$ 0.40 second*,**
3. Pressurizer Pressure - High	$\leq$ 1.15 seconds
4. Thermal Margin/Low Pressure	$\leq$ 0.90 second**
5. Containment Pressure - High	$\leq$ 1.55 seconds
6. Steam Generator Pressure - Low	$\leq$ 1.15 seconds
7. Steam Generator Pressure Difference - High	$\leq$ 1.15 seconds
8. Steam Generator Level - Low	$\leq$ 1.15 seconds
9. Local Power Density - High	$\leq$ 0.40 second*,**

TABLE 3.3-2 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
10. Loss of Component Cooling Water to Reactor Coolant Pumps	Not Applicable
11. Reactor Protection System Logic	Not Applicable
12. Reactor Trip Breakers	Not Applicable
13. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable
14. Reactor Coolant Flow - Low	0.65 second
15. Loss of Load (Turbine Hydraulic Fluid Pressure - Low)	Not Applicable

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

**Based on a resistance temperature detector (RTD) response time of less than or equal to 8.0 seconds where the RTD response time is equivalent to the time interval required for the RTD output to achieve 63.2% of its total change when subjected to a step change in RTD temperature.

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES 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(8)	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(7)	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) - Detector plateau curves shall be obtained, evaluated, and compared to manufacturer's data.

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

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 and with RESPONSE TIMES as shown in Table 3.3-5.

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

TABLE 3.3-3

ENGINEERED SAFETY 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)	17
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	17
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)	1/Bus	1/Bus	1/Bus	1, 2, 3	12
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	2/Bus	2/Bus	2/Bus	1, 2, 3	12
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	12
c. SG 2A - SG 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13*, 14
d. SG Level (2A/2B) - Low	4/SG	2/SG	3/SG	1, 2, 3	13*, 14
e. Feedwater Header SG 2A - SG 2B Differential Pressure	4/SG	2/SG	3/SG	1, 2, 3	13*, 14

**The Auxiliary Feedwater System automatic initiation system shall be completely installed and OPERABLE prior to initial criticality.

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 - High	Containment Pressure - High (SIAS, CIAS, CSAS) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (MSIS) Steam Generator ΔP 1 and 2 (AFAS) Thermal Margin/Low Pressure (RPS)
3.	Steam Generator Level	Steam Generator Level - Low (AFAS, RPS) Steam Generator Level - High (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 Circuit	Containment Pressure - High (SIAS, CIAS, CSAS) Containment Pressure - High (RPS)
2.	Steam Generator Pressure - Low	Steam Generator Pressure - Low (AFAS, MSIS) Steam Generator ΔP 1 and 2 (AFAS) Thermal Margin/Low Pressure (RPS)
3.	Steam Generator Level - Low	Steam Generator Level - Low (RPS) (AFAS) Steam Generator Level - High (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 3.3-4

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	$\leq$ 5.0 psig	$\leq$ 5.10 psig
	c. Pressurizer Pressure - Low	$\geq$ 1736 psia	$\geq$ 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	$\leq$ 9.30 psig	$\leq$ 9.40 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	$\leq$ 5.0 psig	$\leq$ 5.10 psig
	d. Containment Radiation - High	$\leq$ 10 R/hr	$\leq$ 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	$\geq$ 600 psia	$\geq$ 567 psia
	c. Containment Pressure - High	$\leq$ 5.0 psig	$\leq$ 5.10 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)	$\geq$ 3120 volts	$\geq$ 3120 volts
(2) 480 V Emergency Bus Undervoltage (Loss of Voltage)	$\geq$ 360 volts	$\geq$ 360 volts
b. (1) 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	$\geq$ 3848 volts with a 10-second time delay	$\geq$ 3848 volts with a 10-second time delay
(2) 480 V Emergency Bus Undervoltage (Degraded Voltage)	$\geq$ 432 volts	$\geq$ 432 volts
7. AUXILIARY FEEDWATER (AFAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Automatic Actuation Logic	Not Applicable	Not Applicable
c. Steam Generator ΔP-High	$\leq$ 180.0 psid	$\leq$ 187.5 psid
d. SG 2A&2B Level Low	$\geq$ 20.6%	$\geq$ 20.0%
e. Feedwater Header High ΔP	$\leq$ 100.0 psid	$\leq$ 107.5 psid

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual	
a. SIAS	
Safety Injection (ECCS)	Not Applicable
Containment Isolation[(1)]	Not Applicable
Shield Building Ventilation System	Not Applicable
Containment Purge Valve Isolation	Not Applicable
Containment Fan Coolers	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
Iodine Removal	Not Applicable
c. CIAS	
Containment Isolation[(1)]	Not Applicable
Shield Building Ventilation System	Not Applicable
Containment Purge Valve Isolation	Not Applicable
d. MSIS	
Main Steam Isolation	Not Applicable
Feedwater Isolation	Not Applicable
e. RAS	
Containment Sump Recirculation	Not Applicable
f. AFAS	
Auxiliary Feedwater Actuation	Not Applicable
Feedwater Isolation	Not Applicable

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 30.0*/20.0**
b. Containment Isolation(1)	≤ 21.75*/11.75**
c. Shield Building Ventilation System	≤ 26.0*/10.0**
d. Containment Fan Coolers	≤ 24.15*/11.15**
e. Charging Flow	≤ 330.00*/180.00**
3. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 30.0*/20.0**
b. Containment Isolation(1)	≤ 21.75*/11.75**
c. Shield Building Ventilation System	≤ 26.0*/10.0**
d. Containment Fan Coolers	≤ 24.15*/11.15**
e. Feedwater Isolation	≤ 5.35*/5.35**
f. Main Steam Isolation	≤ 6.75*/6.75**
4. Containment Pressure--High-High	
a. Containment Spray/Iodine Removal	≤ 25.65*/11.15**
5. Containment Radiation-High	
a. Containment Isolation(1)	≤ 26.75*/16.75**
b. Shield Building Ventilation System	≤ 32.75*/16.75**
6. Steam Generator Pressure-Low	
a. Feedwater Isolation	≤ 5.35/5.35**
b. Main Steam Isolation	≤ 6.75/6.75**
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	≤ 111.15*/101.15**
8. 4.16 kV Emergency Bus Undervoltage (Loss of Voltage)	
a. Loss of Power (4.16 kV)	≤ 14
b. Loss of Power (480 V)	≤ 14
9. 4.16 kV Emergency Bus Undervoltage (Degraded Voltage)	
a. Loss of Power (4.16 kV)	≤ 12
b. Loss of Power (480 V)	≤ 22

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
10. Steam Generator Level-Low	
a. Auxiliary Feedwater	$\leq$ 120*/120**
b. Feedwater Isolation	$\leq$ 5.35*/5.35**
11. Feedwater Header ΔP	
a. Auxiliary Feedwater	$\leq$ 120*/120**
b. Feedwater Isolation	$\leq$ 5.35*/5.35**
12. Steam Generator ΔP	
a. Auxiliary Feedwater	$\leq$ 120*/120**
b. Feedwater Isolation	$\leq$ 5.35*/5.35**

NOTE: Response time for Motor-Driven and Steam-Driven Auxiliary Feedwater Pumps on all AFAS signal starts	$\leq$ 120.0

TABLE NOTATION

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

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

(1)Containment Isolation response time is applicable to the valves specified in Specification 3.6.3.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTAION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. SAFETY INJECTION (SIAS)				
a. Manual (Trip Buttons)	N.A.	N.A.	R	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), SA(2)	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), SA(2)	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), SA(2)	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), SA(2)	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), SA(2)	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 and Pressure (A/B)-Low and ΔP (A/B) - High	S	R	M	1, 2, 3
c. SG Level (A/B) - Low and No Pressure - Low Trip (A/B)	S	R	M	1, 2, 3
d. Automatic Actuation Logic	N.A.	N.A.	M(1), SA(2)	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) A subgroup relay test shall be performed which shall include the energization/de-energization of each subgroup relay (solid-state component) and verification of the OPERABILITY of each subgroup relay (solid-state component).

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	*	$\leq$ 20 mR/hr	10^{-1} - 10^{4} mR/hr	22
b. Containment Isolation	3	6	$\leq$ 90 mR/hr	1 - 10^{7} mR/hr	25
c. Control Room Isolation	1 per intake	ALL MODES	$\leq$ 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 Specification 3.3.3.10.

****Monitors shall be completely installed and OPERABLE prior to exceeding 5% of RATED THERMAL POWER.

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	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 Specification 3.3.3.10.

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

****Monitors shall be completely installed and OPERABLE prior to exceeding 5% of RATED THERMAL POWER.

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, comply with the ACTION requirements of Specification 3.6.6.1.

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.

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 The incore detection system shall be OPERABLE with:

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

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

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

APPLICABILITY: When the incore detection system is used for:

a. Recalibration of the excore axial flux offset detection system,

b. Monitoring the AZIMUTHAL POWER TILT,

c. Calibration of the power level neutron flux channels, or

d. Monitoring the linear heat rate.

ACTION:

a. With the incore detection system inoperable, do not use the system for the above applicable monitoring or calibration functions.

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 24 hours prior to its use and at least once per 7 days thereafter when required for:

1. Recalibration of the excore axial flux offset detection system,

2. Monitoring the linear heat rate pursuant to Specification 4.2.1.3,

SURVEILLANCE REQUIREMENTS (Continued)

3. Monitoring the AZIMUTHAL POWER TILT, or

4. Calibration of the Power Level Neutron Flux Channels.

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

SEISMIC INSTRUMENTATION*

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE seismic monitoring channels less than required by Table 3.3-7, restore the inoperable channel(s) to OPERABLE status within 30 days.

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.3.3.2 Each of the above seismic monitoring instruments actuated during a seismic event (greater than or equal to 0.01g) shall be restored to OPERABLE status within 24 hours and a CHANNEL CALIBRATION performed within 5 days. Data shall be retrieved from actuated instruments and analyzed to determine the magnitude of the vibratory ground motion. A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.

*The Seismic Instrumentation System is shared between St. Lucie - Unit 1 and St. Lucie - Unit 2.

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION
(Instrumentation located in St. Lucie Unit 1)

INSTRUMENT CHANNEL		SENSOR LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	STRONG MOTION TRIAXIAL ACCELEROGRAPHS			
a.	SMR-42-1	R.B. Elev. 23.0'	0-1 g	1*
b.	SMR-42-2	R.B. Elev. 62.0'	0-1 g	1
c.	SMR-42-3	R.A.B. Elev. -0.5'	0-1 g	1
d.	SMR-42-4	R.A.B. Elev. 43.0'	0-1 g	1
2.	PEAK RECORDING ACCELEROGRAPHS			
a.	SMR-42-6	R.B. Piping from S.I.T.1A2-c Elev. 46' 10 9/16"	0-2 g	1
b.	SMR-42-7	R.B. Equipment on S.I.T.1A2	0-2 g	1
c.	SMR-42-8	R.A.B.-Sh. Dn. Ht. XCHR Supports	0-2 g	1
3.	PEAK SHOCK RECORDERS			
a.	SMR-42-9	R.B. Elev. 23.0'	-	1
b.	SMR-42-10	R.B. M.S. Pipe Restraints - S.G.1B1	-	1
4.	EARTHQUAKE FORCE MONITOR			
a.	SMI-42-11	Control Room	0-0.2 g	1
5.	SEISMIC SWITCH			
a.	SMS-42-12	R.B. Elev. 23.0'	-	1*

*With St. Lucie Unit 2 reactor control room alarm

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS
(Instrumentation located in St. Lucie Unit 1)

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. STRONG MOTION TRIAXIAL ACCELEROGRAPHS			
a. SMR-42-1	M*	R	SA
b. SMR-42-2	M*	R	SA
c. SMR-42-3	M*	R	SA
d. SMR-42-4	M*	R	SA
e. SMR-42-5	M*	R	SA
2. PEAK RECORDING ACCELEROGRAPHS			
a. SMR-42-6	NA	R	NA
b. SMR-42-7	NA	R	NA
c. SMR-42-8	NA	R	NA
3. PEAK SHOCK RECORDERS			
a. SMR-42-9	M	R	SA
b. SMR-42-10	M	R	SA
4. EARTHQUAKE FORCE MONITOR			
a. SMI-42-11	M	R	SA
5. SEISMIC SWITCH			
a. SMS-42-12	NA	R	SA

*Except seismic trigger

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION*

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

a. With the number of OPERABLE meteorological monitoring channels less than required by Table 3.3-8, suspend all release of gaseous radioactive material from the radwaste gas decay tanks until the inoperable channel(s) is restored to OPERABLE status.

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

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

SURVEILLANCE REQUIREMENTS

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

*The Meteorological Instrumentation system is shared between St. Lucie - Unit 1 and St. Lucie - Unit 2.

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT & ELEVATION	INSTRUMENT MINIMUM ACCURACY	MINIMUM CHANNELS OPERABLE
1. WINDSPEED		
a) Nominal Elev (10 meters)	± 0.5 *mph	1[A]
b) Nominal Elev (57.9 meters)	± 0.5 *mph	N.A.
2. WIND DIRECTION		
a) Nominal Elev (10 meters)	± 5°	1[B]
b) Nominal Elev (57.9 meters)	± 5°	N.A.
3. AIR TEMPERATURE (Delta T)		
a) Nominal Elev (10 meters)	± 0.1°C**	1[C]
b) Nominal Elev (57.9 meters)	± 0.1°C**	1[C]
c) Nominal Elev (33.5 meters)	± 0.1°C**	N.A.

*Starting speed of anemometer shall be < 1 mph.

**ΔT measurement channels only.

[A]The 57.9-meter channel may be substituted for the 10-meter wind speed for up to 30 days in the event the 10-meter channel is inoperable. Wind speed data from the 57.9-meter elevation should be adjusted using the wind speed power law:

$$S_{10\ meters} = S_{57.9\ meters} \quad (0.1727)^n$$

where:

S = wind speed in mph
n = 0.25 for Pasquill Vertical Stability Classes A,B,C,D.
n = 0.50 for Pasquill Vertical Stability Classes E,F,G.
1.727×10^{-1} = constant = 10 meters/57.9 meters

[B]The 57.9-meter channel may be substituted for the 10-meter wind direction channel for up to 30 days in the event the 10-meter channel is inoperable.

[C]The 33.5-meter channel may be substituted for one of the 10-meter or 57.9-meter temperature channels for up to 30 days if one of the channels is inoperable. The data should always be normalized to °C/100 meters to determine the vertical stability class.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a) Nominal Elev (10 meters)	D	SA
b) Nominal Elev (57.9 meters)	D*	SA*
2. WIND DIRECTION		
a) Nominal Elev (10 meters)	D	SA
b) Nominal Elev (57.9 meters)	D*	SA*
3. AIR TEMPERATURE (DELTA T)		
a) Nominal Elev (10 meters)	D	SA
b) Nominal Elev (57.9 meters)	D	SA
c) Nominal Elev (33.5 meters)	D*	SA*

*Required only if these channels are being substituted for one of the Minimum Channels Operable as per Table 3.3-8.

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	$2 \times 10^{-8}\%$ - 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	2/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

INSTRUMENTATION

ACCIDENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

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} (Narrow 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

*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 discharge piping temperature at least once every 12 hours.

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

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} (Narrow 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
16. Containment Sump Water Level (Narrow Range)	M	R
16. Containment Water Level (Wide Range)	M	R
17. Incore Thermocouples	M	R

FIRE DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

a. With any, but not more than one-half the total in any fire zone, Function A fire detection instruments shown in Table 3.3-11 inoperable, restore the inoperable instrument(s) to OPERABLE status within 14 days or within the next 1 hour establish a fire watch patrol to inspect the zone(s) with the inoperable instrument(s) at least once per hour, unless the instrument(s) is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in Specification 4.6.1.5.

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

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

SURVEILLANCE REQUIREMENTS

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

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

4.3.3.7.3 The nonsupervised circuits, associated with detector alarms, between the instrument and the control room shall be demonstrated OPERABLE at least once per 31 days.

TABLE 3.3-11

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS*		
	HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
REACTOR AUXILIARY BUILDING			
ZONE-1A REACTOR AUX. BLDG. EL. 0.50			6/0
ZONE-2A REACTOR AUX. BLDG. EL. 0.50			4/0
ZONE-3A REACTOR AUX. BLDG. EL. 19.50			6/0
ZONE-4 REACTOR AUX. BLDG. EL. 19.50		2/0	5/0
ZONE-5A REACTOR AUX. BLDG. EL. 19.50			8/0
ZONE-6A REACTOR AUX. BLDG. EL. 43.00			5/0
ZONE-7A REACTOR AUX. BLDG. EL. 43.00			7/0
ZONE-8A REACTOR AUX. BLDG. EL. 62.00		1/0	6/0
ZONE-9A REACTOR AUX. BLDG. EL. 43.00			2/0
ZONE-10A REACTOR AUX. BLDG. EL. 43.00			2/0
ZONE-1B REACTOR AUX. BLDG. EL. 0.50			6/0
ZONE-2B REACTOR AUX. BLDG. EL. 0.50			5/0
ZONE-3B REACTOR AUX. BLDG. EL. 19.50			6/0
ZONE-4B HSCP-1 REC. AUX. BLDG. EL. 43.00			1/0
ZONE-5B REACTOR AUX. BLDG. EL. 19.50			6/0
ZONE-6B REACTOR AUX. BLDG. EL. 43.00			4/0
ZONE-7B REACTOR AUX. BLDG. EL. 43.00			6/0
ZONE-8B REACTOR AUX. BLDG. EL. 62.00			5/0
ZONE-9B REACTOR AUX. BLDG. EL. 43.00			2/0
ZONE-10B REACTOR AUX. BLDG. EL. 43.00			2/0
ZONE-1F FAN ROOM EL. 43.00	0/2		
ZONE-2F CABLE LOFT EL. 19.50	0/26		
ZONE-3F IODINE REMOVAL/WASTE GAS/ HALLWAYS EL. 0.50	0/15		
ZONE-4F B ELECTRICAL PENETRATION ROOM EL. 19.50	0/2		
ZONE-5F A ELECTRICAL PENETRATION ROOM EL. 19.50	0/1		
ZONE-6F CABLE SPREADING ROOM EL. 43.00	0/9		
FUEL HANDLING BUILDING			
ZONE-20A FUEL HANDLING BLDG. EL. 19.50			1/0
ZONE-21A FUEL HANDLING BLDG. EL. 48.00			3/0
ZONE-20B FUEL HANDLING BLDG. EL. 19.50			1/0
ZONE-21B FUEL HANDLING BLDG. EL. 48.00			2/0
DIESEL GENERATOR BUILDING			
ZONE-22A DIESEL GEN. BLDG.	2/2		2/0
ZONE-22B DIESEL GEN. BLDG.	2/2		2/0

TABLE 3.3-11 (Continued)

FIRE DETECTION INSTRUMENTS

INSTRUMENT LOCATION	TOTAL NUMBER OF INSTRUMENTS* HEAT (x/y)	FLAME (x/y)	SMOKE (x/y)
SAFETY RELATED PUMPS			
ZONE-17A COMPONENT COOLING AREA			4/0
ZONE-18A INTAKE COOLING WATER PUMP AREA	1/0		
ZONE-19A STEAM TRESTLE AREA-AUX. FEEDWATER PUMP			2/0
ZONE-17B COMPONENT COOLING AREA			2/0
ZONE-18B INTAKE COOLING WATER PUMP AREA	1/0		
ZONE-19B STEAM TRESTLE AREA-AUX. FEEDWATER PUMP			2/0
TURBINE BUILDING/SWITCHGEAR ROOM			
ZONE-16A TURBINE BLDG. SWITCHGEAR ROOM			3/0
ZONE-16B TURBINE BLDG. SWITCHGEAR ROOM			3/0
CONTAINMENT#			
ZONE-13A REACTOR TUNNEL BELOW EL. 18.00			2/0
ZONE-14A REACTOR EL. 18.00			5/0
ZONE-15A REACTOR EL. 45.00	2/0		4/0
ZONE-13B REACTOR TUNNEL BELOW EL. 18.00			1/0
ZONE-14B REACTOR EL. 18.00			5/0
ZONE-15B REACTOR EL. 45.00	2/0		5/0
CABLE SPREADING			
ZONE-11A ELECT. PENET. REC. (ANNULUS)			1/0
ZONE-12A ELECT. PENET. REC. AUX. BLDG. EL. 19.50			3/0
ZONE-11B ELECT. PENET. REC. (ANNULUS)			1/0
ZONE-12B ELECT. PENET. REC. AUX. BLDG. EL. 19.50			4/0

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

y is number of Function B (actuation of fire suppression systems and early warning and notification) instruments.

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

INSTRUMENTATION

LOOSE-PART DETECTION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.8 The loose-part detection system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. a CHANNEL FUNCTIONAL TEST at least once per 31 days, and

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

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

b. With less than the minimum number of radioactive liquid effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-12. If the inoperable instruments are not returned to operable status within 30 days, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4 and 6.9.1.9 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-12

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	ACTION
1. GROSS RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE		
a. Liquid Radwaste Effluent Line	1	28
b. Steam Generator Blowdown Effluent Line	1/SG	29
2. FLOW RATE MEASUREMENT DEVICES		
a. Liquid Radwaste Effluent Line	NA	30
b. Discharge Canal	NA	30
c. Steam Generator Blowdown Effluent Lines	NA	30

TABLE 3.3-12 (Continued)

TABLE NOTATION

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

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

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge line valving (one performs, one verifies);

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 29 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, effluent releases via this pathway may continue for up to 30 days provided grab samples are analyzed for gross radioactivity (beta or gamma) at a limit of detection of at least 2×10^{-7} microcuries/gram:

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

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

ACTION 30 - Minimum system design flow of required running pumps shall be utilized for MPC calculations for discharge canal flow and maximum system design flow shall be utilized for MPC calculations for effluent line flow.

TABLE 4.3-8

RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. GROSS BETA OR GAMMA RADIOACTIVITY MONITORS PROVIDING ALARM AND AUTOMATIC TERMINATION OF RELEASE				
a. Liquid Radwaste Effluent Line	D	P	R(2)	Q(1)
b. Steam Generator Blowdown Effluent Line	D	M	R(2)	Q(1)
2. FLOW RATE MEASUREMENT DEVICES				
a. Liquid Radwaste Effluent Line	D	N.A.	R	Q
b. Steam Generator Blowdown Effluent Line	D	N.A.	R	Q

TABLE 4.3-8 (Continued)

TABLE NOTATION

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

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

(2) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards traceable to the National Bureau of Standards or using standards that have been calibrated against standards certified by the NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities that are typical of normal plant operation. For subsequent CHANNEL CALIBRATION, button sources that have been related to the initial calibration may be used.

INSTRUMENTATION

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-13

ACTION:

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

b. With less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels OPERABLE, take the ACTION shown in Table 3.3-13. If the inoperable instruments are not returned to operable status within 30 days, explain in the next Semiannual Radioactive Effluent Release Report why the inoperability was not corrected in a timely manner.

c. The provisions of Specifications 3.0.3, 3.0.4, and 6.9.1.9 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-13

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
1. WASTE GAS DECAY TANKS			
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	1	*	35
2. WASTE GAS DECAY TANKS EXPLOSIVE GAS MONITORING SYSTEM			
a. Oxygen Monitors	1	**	39
3. CONDENSER EVACUATION SYSTEM			
a. Noble Gas Activity Monitor	1	***	37
4. PLANT VENT SYSTEM			
a. Noble Gas Activity Monitor (Low Range)	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate	NA	*	38
e. Sampler Flow Rate Monitor	1	*	36

TABLE 3.3-13 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION

INSTRUMENT	MINIMUM CHANNELS OPERABLE	APPLICABILITY	ACTION
5. FUEL STORAGE AREA VENTILATION SYSTEM			
a. Noble Gas Activity Monitor (Low Range)	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	NA	*	38
e. Sampler Flow Rate Monitor	1	*	36
6. LAUNDRY AREA VENTILATION SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate	NA	*	38
e. Sampler Flow Rate Monitor	1	*	36
7. STEAM GENERATOR BLOWDOWN BUILDING VENT SYSTEM			
a. Noble Gas Activity Monitor	1	*	37
b. Iodine Sampler	1	*	40
c. Particulate Sampler	1	*	40
d. Flow Rate Monitor	NA	*	38
e. Sampler Flow Rate Monitor	1	*	36

TABLE 3.3-13 (Continued)

TABLE NOTATION

* At all times.

** During waste gas system operation.

***At all times when air ejector exhaust is not directed to plant vent.

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

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

b. At least two technically qualified members of the Facility Staff independently verify the release rate calculations and discharge valve lineup (one performs, one verifies).

Otherwise, suspend release of radioactive effluents via this pathway.

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

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

ACTION 38 - Maximum system flows shall be utilized in the determination of the instantaneous release monitor alarm setpoint.

ACTION 39 - With the number of channnels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, operation of this system may continue for up to 30 days provided samples of O_2 are analyzed by the lab gas partitioner at least once per 24 hours.

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

TABLE 4.3-9

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. WASTE GAS DECAY TANKS					
a. Noble Gas Activity Monitor - Providing Alarm and Automatic Termination of Release	P	P	R(3)	Q(1)	*
2. WASTE GAS DECAY TANKS EXPLOSIVE GAS MONITORING SYSTEM					
a. Oxygen Monitor	D	N.A.	Q(5)	M	**
b. Oxygen Monitor (alternate)	D	N.A.	Q(5)	M	**
3. CONDENSER EVACUATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	***
4. PLANT VENT SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

TABLE 4.3-9 (Continued)

RADIOACTIVE GASEOUS EFFLUENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	SOURCE CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. FUEL STORAGE AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A.	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
6. LAUNDRY AREA VENTILATION SYSTEM					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*
7. STEAM GENERATOR BLOWDOWN BUILDING VENT					
a. Noble Gas Activity Monitor	D	M	R(3)	Q(2)	*
b. Iodine Sampler	W	N.A.	N.A.	N.A	*
c. Particulate Sampler	W	N.A.	N.A.	N.A.	*
d. Sampler Flow Rate Monitor	D	N.A.	R	N.A.	*

TABLE 4.3-9 (Continued)

TABLE NOTATION

* At all times other than when the line is valved out and locked.

** During waste gas holdup system operation.

*** At all times when air ejector exhaust is not directed to plant vent.

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

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

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

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

(3) The initial CHANNEL CALIBRATION for radioactivity measurement instrumentation shall be performed using one or more of the reference standards traceable to the National Bureau of Standards or using standards that have been calibrated against standards certified by the NBS. These standards should permit calibrating the system over its intended range of energy and rate capabilities that are typical of normal plant operation. For subsequent CHANNEL CALIBRATION, button sources that have been related to the initial calibration may be used, at intervals of at least once per 18 months.

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

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

INSTRUMENTATION

3/4.3.4 TURBINE OVERSPEED PROTECTION

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

b. With the above required turbine overspeed protection system otherwise inoperable, within 6 hours isolate the turbine from the steam supply.

SURVEILLANCE REQUIREMENTS

4.3.4.1 The provisions of Specification 4.0.4 are not applicable.

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

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

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

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

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

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

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*With any main steam line isolation valve and/or any main steam line isolation valve bypass valve not fully closed.

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 Exceptions 3.10.3 and 3.10.6.

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 all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required Reactor Coolant loop to operation.

SURVEILLANCE REQUIREMENTS

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

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

4.4.1.2.3 The required steam generator(s) shall be determined OPERABLE verifying the secondary side water level to be $\geq$ 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 dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

**The requirements of Specification 3.4.1.2 may be suspended for natural circulation training prior to initial criticality.

REACTOR COOLANT SYSTEM

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 corective 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 all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required coolant loop to operation.

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

**A Reactor Coolant pump shall not be started with one or more of the Reactor Coolant System cold leg temperatures less than or equal to 320°F during heatup or 280°F during cooldown unless the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

***The requirements of Specification 3.4.1.3 may be suspended for natural circulation training prior to initial criticality.

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.

REACTOR COOLANT SYSTEM

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 all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required shutdown cooling loop 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.

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

#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 one or more of the Reactor Coolant System cold leg temperatures less than or equal to 320°F during heatup or 280°F during cooldown unless the secondary water temperature of each steam generator is less than 100°F above each of the Reactor Coolant System cold leg temperatures.

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 all operations involving a reduction in boron concentration of the Reactor Coolant System 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 dilution of the Reactor Coolant System boron concentration, and (2) core outlet temperature is maintained at least 10°F below saturation temperature.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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 2500 psia ± 1%.*

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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 65% 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.

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:

SURVEILLANCE REQUIREMENTS (Continued)

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

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

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

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

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

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

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

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

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

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

a. The first inservice inspection 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.

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.

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

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

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

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

TABLE 4.4-2

STEAM GENERATOR TUBE INSPECTION

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

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

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

a. A containment atmosphere particulate radioactivity monitoring system,

b. The reactor cavity sump level and flow monitoring system, and

c. A containment atmosphere gaseous radioactivity monitoring system.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

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

REACTOR COOLANT SYSTEM

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 72 hours or more and if leakage testing has not been performed in the previous 9 months,

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

d. 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: At all times.

ACTION:

MODES 1, 2, 3, and 4:

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

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

At All Other Times:

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

SURVEILLANCE REQUIREMENTS

4.4.7 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

REACTOR COOLANT SYSTEM

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

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

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

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

ACTION: (Continued)

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

d. 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. A REPORTABLE OCCURRENCE shall be prepared and submitted to the Commission pursuant to Specification 6.9.1. This report shall contain the results of the specific activity analyses together with the following information:

1. Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded,
2. Fuel burnup by core region,
3. Clean-up flow history starting 48 hours prior to the first sample in which the limit was exceeded,
4. History of degassing operation, if any, starting 48 hours prior to the first sample in which the limit was exceeded, and
5. The time duration when the specific activity of the primary coolant exceeded 1 microcurie/gram DOSE EQUIVALENT I-131.

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.

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 mCi/gram, DOSE EQUIVALENT I-131 or 100/$\bar{E}$ mCi/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.

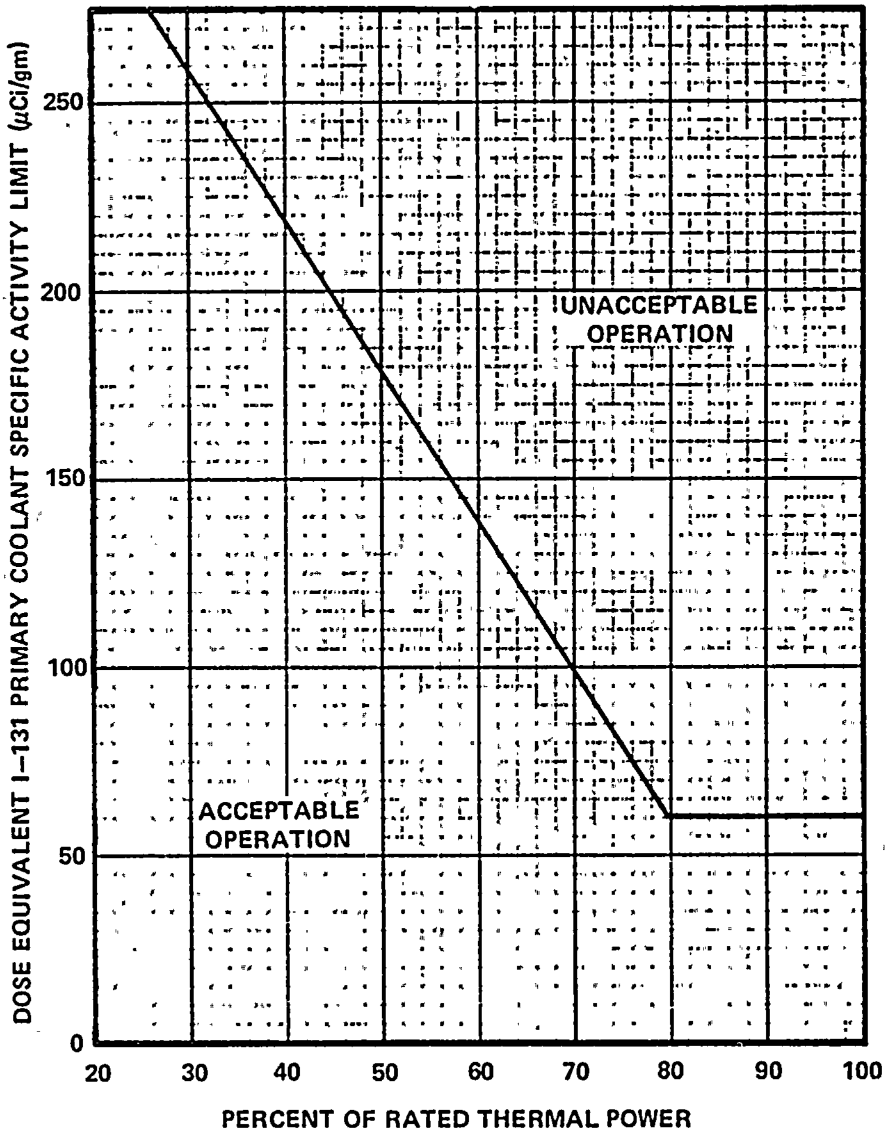

FIGURE 3.4-1

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

a. A maximum heatup of 10°F in any 1-hour period with RCS cold leg temperature less than 92°F. A maximum heatup of 30°F in any 1-hour period with RCS cold leg temperature greater than 92°F but less than 118°F. A maximum heatup of 50°F in any 1-hour period with RCS cold leg temperature greater than 118°F but less than 370°F. A maximum heatup of 100°F in any 1-hour period with RCS cold leg temperature greater than 370°F.

b. A maximum cooldown of 10°F in any 1-hour period with RCS cold leg temperature less than 75°F. A maximum cooldown of 30°F in any 1-hour period with RCS cold leg temperature greater than 75°F but less than 86°F. A maximum cooldown of 50°F in any 1-hour period with RCS cold leg temperature greater than 86°F but less than 97°F. A maximum cooldown of 100°F in any 1-hour period with RCS cold leg temperature greater than 97°F.

c. A maximum temperature change of less than or equal to 10°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 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, at the intervals required by 10 CFR 50 Appendix H in accordance with the schedule in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3, and 3.4-4.

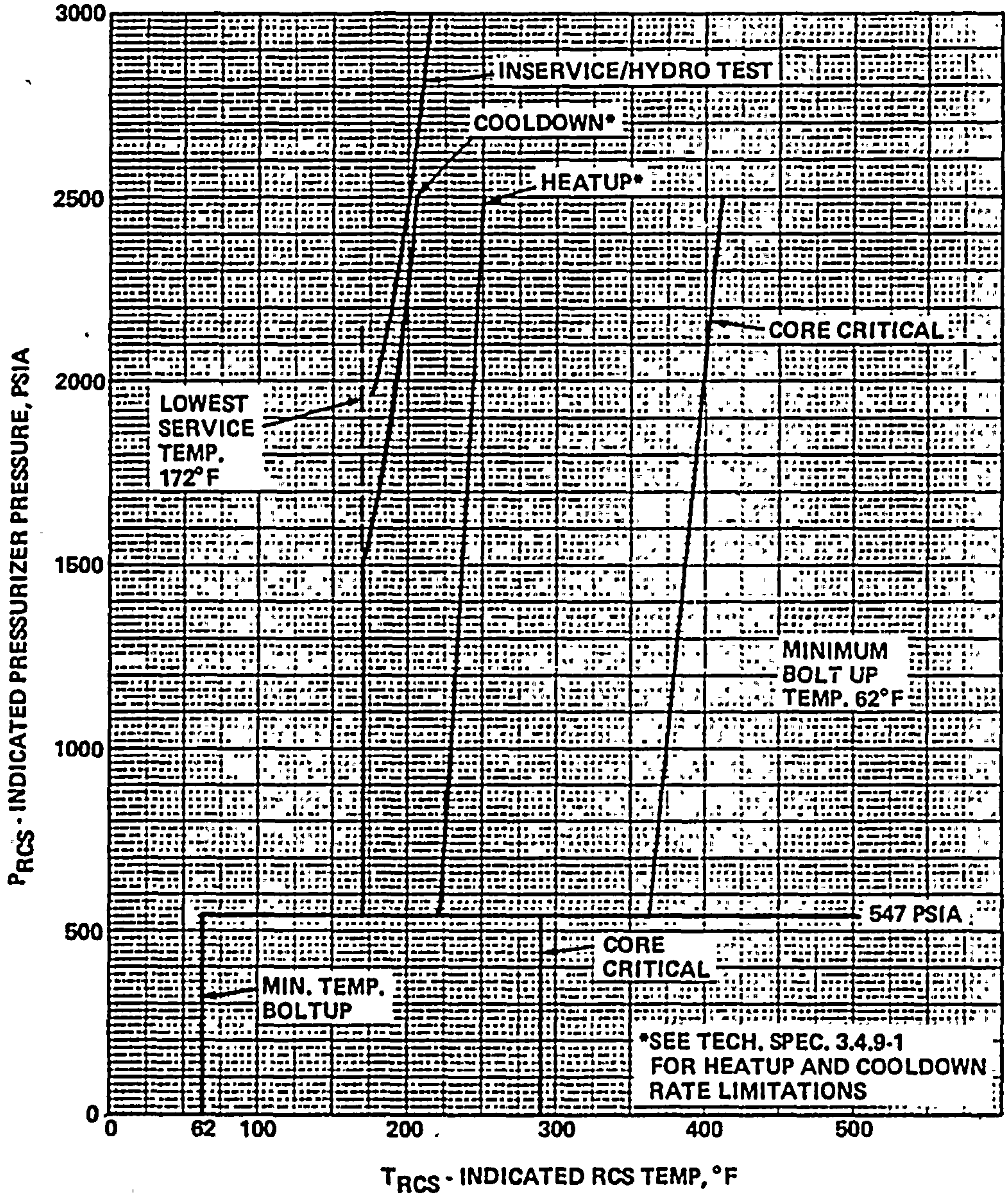

FIGURE 3.4-2

REACTOR COOLANT SYSTEM
PRESSURE TEMPERATURE LIMITATIONS
0 TO 2 YEARS OF OPERATION

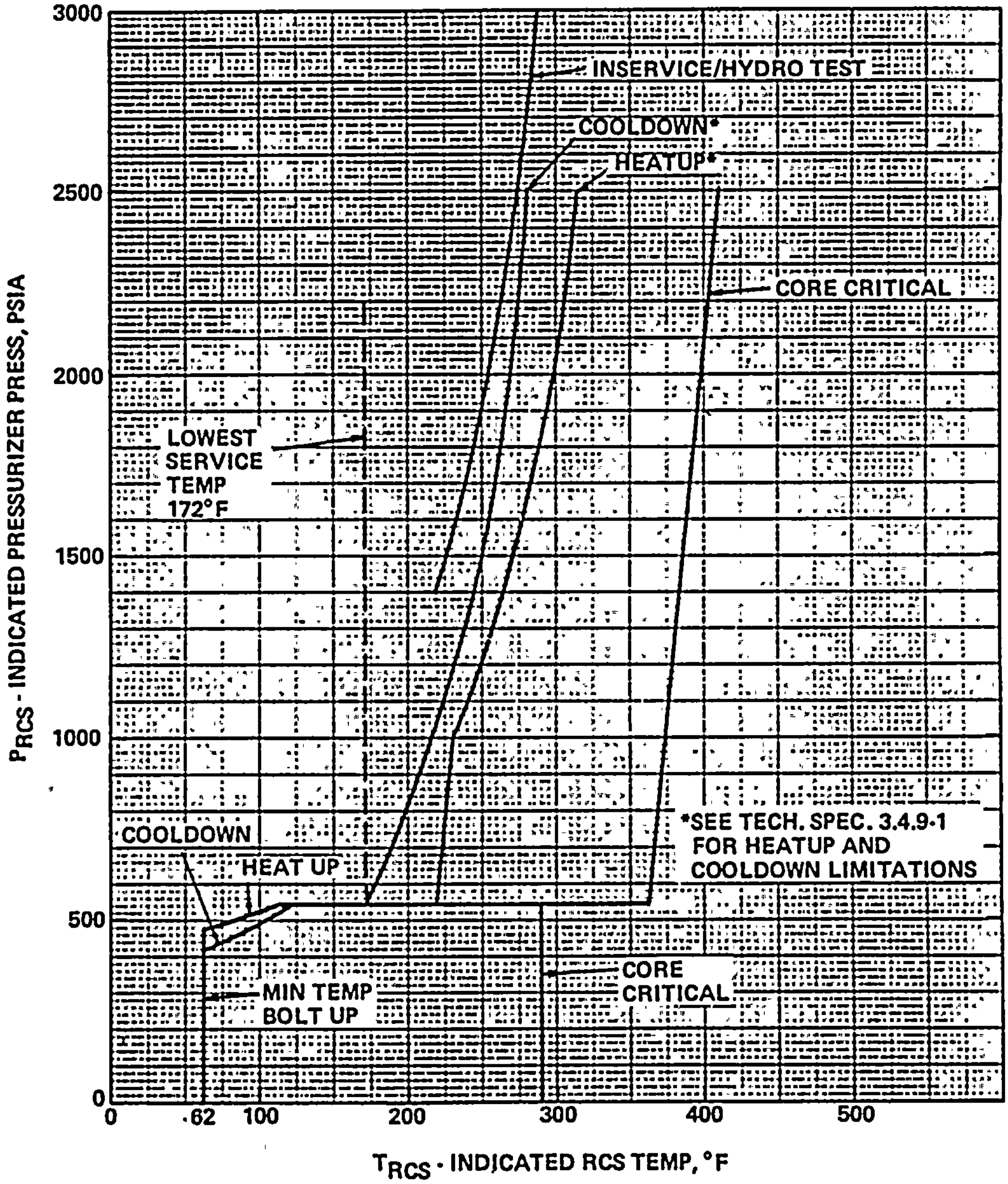

FIGURE 3.4-3

REACTOR COOLANT SYSTEM PRESSURE TEMPERATURE LIMITATIONS 2 TO 10 YEARS OF OPERATION

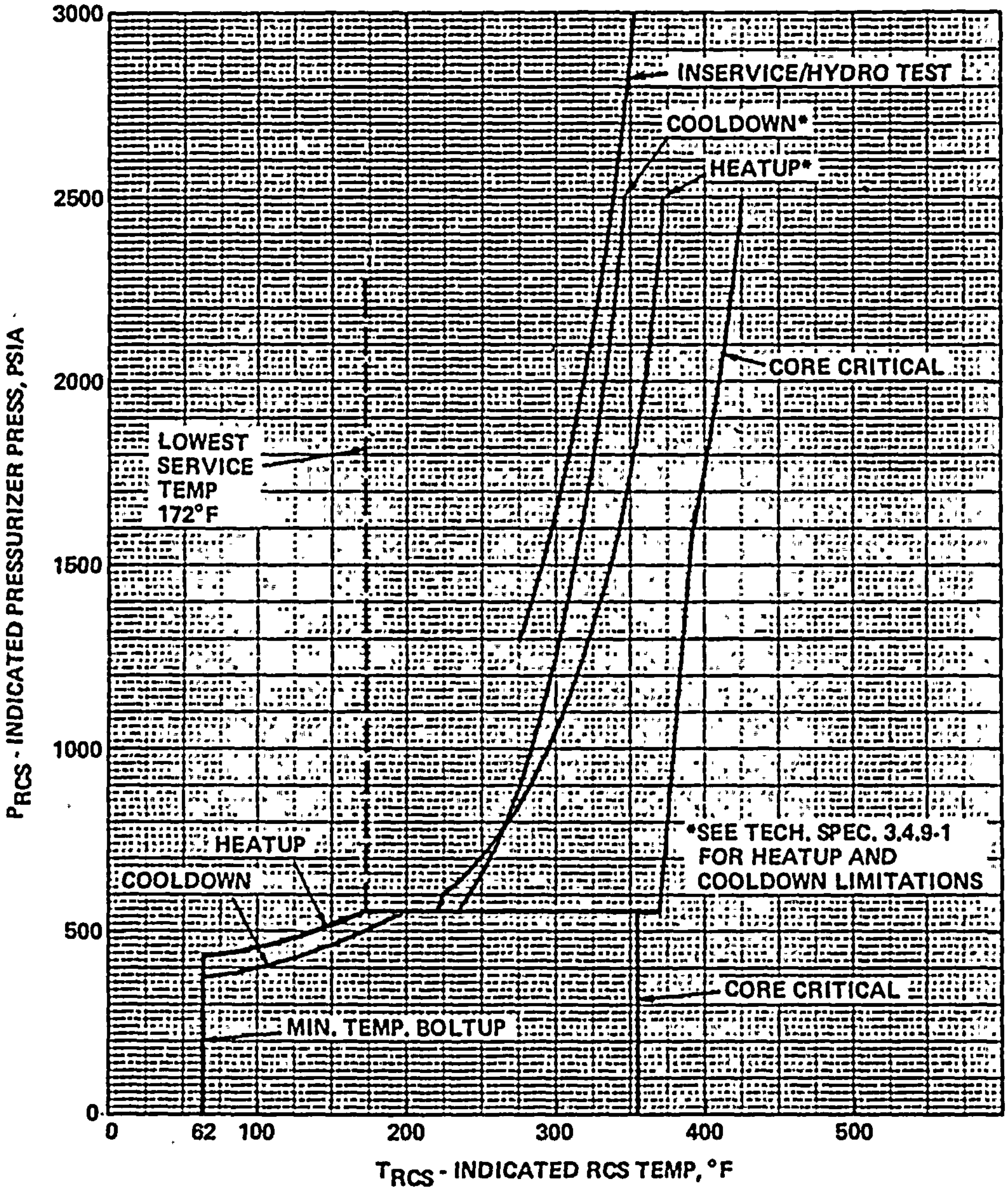

FIGURE 3.4-4

REACTOR COOLANT SYSTEM
PRESSURE TEMPERATURE LIMITATIONS
10 TO 40 YEARS OF OPERATION

TABLE 4.4-5

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM - WITHDRAWAL SCHEDULE

CAPSULE NUMER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	83°	1.15	1.0
2	97°	1.15	24.0
3	104°	1.15	STANDBY
4	263°	1.15	12.0
5	277°	1.15	STANDBY
6	284°	1.15	STANDBY

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

a. Two power-operated relief valves (PORVs), one with a lift setting of less than or equal to 460 psia, and one with a lift setting of less than or equal to 490 psia, or,

b. The Reactor Coolant System depressurized with an RCS vent of greater than or equal to 3.58 square inches.

APPLICABILITY: MODE 4 when the temperature of any RCS cold leg is less than or equal 280°F during cooldown and 320°F during heatup, MODE 5, and MODE 6 with the reactor vessel head on.

ACTION:

a. With one PORV inoperable, restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through a greater than or equal to 3.58 square inch vent(s) within the next 8 hours.

b. With both PORVs inoperable, depressurize and vent the RCS through a greater than or equal to 3.58 square inch vent(s) within 8 hours.

c. In the event either the PORVs 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 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 Specification 4.0.5, operating the valve 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.

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.

*The Reactor Coolant System vents shall be completely installed and OPERABLE prior to exceeding 5% of RATED THERMAL POWER.

REACTOR COOLANT SYSTEM

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

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 570 and 650 psig.

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

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in at least HOT STANDBY within 1 hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

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

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

*With pressurizer pressure greater than or equal to 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 1540 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 1540 cubic feet with a boron concentration of between 1720 and 2100 ppm of boron. In MODE 4 with pressurizer pressure less than 276 psia, the safety injection tanks may be isolated.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the safety injection tank solution.

c. At least once per 31 days when the RCS pressure is above 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.

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

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

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

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 325°F

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE high pressure safety injection pump,

b. One OPERABLE low pressure safety injection pump, and

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

d. One OPERABLE charging pump.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

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

*With pressurizer pressure greater than or equal to 1750 psia.

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. Of the areas affected within containment at the completion of containment 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.

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 ± 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 ± 0.1 gallons of 120 ± 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 on a Sump Recirculation Actuation Test Signal, the containment sump isolation valves open and the recirculation valve to the refueling water tank closed.

g. By verifying that each of the following pumps develops the specified total developed head on recirculation flow when tested pursuant to Specification 4.0.5:

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.	HVC 3616/3617	a.	HCV 3615
b.	HVC 3626/3627	b.	HCV 3625
c.	HVC 3636/3637	c.	HCV 3635
d.	HVC 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.

3/4.5.3 ECCS SUBSYSTEMS - T_{avg} LESS THAN 325°F

LIMITING CONDITION FOR OPERATION

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

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

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

APPLICABILITY: MODES 3* and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

*With pressurizer pressure less than 1750 psia.

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 as provided in Table 3.6-2 of Specification 3.6.4.

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

c. After each closing of each penetration subject to Type B testing, except containment air locks, if opened following a Type A or B test, by leak rate testing the seal with gas at P_a 43.4 psig and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Specification 4.6.1.2d. for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

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*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 to:

a. An overall integrated leakage rate of:

1. Less than or equal to L_a, 0.50 percent by weight of the containment air per 24 hours at P_a, 43.4 psig, or

2. Less than or equal to L_t, 0.35 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, 21.7 psig.

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

c. A combined bypass leakage rate of less of than or equal 0.12 L_a for all penetrations identified in Table 3.6-1 as secondary containment bypass leakage paths when pressurized to P_a.

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

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding 0.75 L_a or 0.75 L_t, as applicable, or (b) with the measured combined leakage rate for all penetrations and valves subject to Types B and C tests exceeding 0.60 L_a, or (c) with the combined bypass leakage rate exceeding 0.12 L_a, restore the overall integrated leakage rate to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable, and the combined leakage rate for all penetrations and valves subject to Type B and C tests to less than or equal to 0.60 L_a, and the bypass leakage rate to less than or equal to 0.12 L_a prior to increasing the Reactor Coolant System temperature above 200°F.

SURVEILLANCE REQUIREMENTS

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

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 ± 10 month intervals during

SURVEILLANCE REQUIREMENTS (Continued)

shutdown at either P_a, 43.4 psig or at P_t, 21.7 psig during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

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

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

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

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

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage rate at P_a, 43.4 psig or P_t, 21.7 psig.

d. Type B and C tests shall be conducted with gas at P_a, 43.4 psig at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Purge supply and exhaust isolation valves with resilient material seals.

e. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.7.3 and 4.6.1.7.4.

SURVEILLANCE REQUIREMENTS (Continued)

f. The combined bypass leakage rate shall be determined to be less than or equal to 0.12 L_a by applicable Type B and C tests at least once per 24 months except for penetrations which are not individually testable; penetrations not individually testable shall be determined to have no detectable leakage when tested with soap bubbles while the containment is pressurized to P_a, 43.4 psig during each Type A test.

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

h. The provisions of Specification 4.0.2 are not applicable.

TABLE 3.6-1

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number/Type	Location to Containment	Service	Test Type*
7	Makeup Water	I-HCV-15-1 Globe I-V-15-328 Check	Outside Inside	Primary Makeup Water	BYPASS/ TYPE C
8	Station Air	I-V-18-794 Globe I-V-18-796 Globe I-V-18-797 Globe	Outside Inside Outside	Station Air Supply	BYPASS/ TYPE C
9	Instrument Air	I-HCV-18-1 Globe I-V-18-195 Check	Outside Inside	Instrument Air Supply	BYPASS/ TYPE C
10	Containment Purge	I-FCV-25-5 B'FLY I-FCV-25-4 B'FLY	Annulus Inside	Containment Purge Exhaust	TYPE C
11	Containment Purge	I-FCV-25-2 B'FLY I-FCV-25-3 B'FLY	Annulus Inside	Containment Purge Supply	TYPE C
14	Waste Management	V-6741 Globe V-6792 Check	Outside Inside	N_2 Supply to Safety Inj. Tanks	BYPASS/ TYPE C
23	Component Cooling	I-HCV-14-7 B'FLY I-HCV-14-1 B'FLY	Outside Inside	RC Pump Cooling Water Supply	BYPASS/ TYPE C
24	Component Cooling	I-HCV-14-6 B'FLY I-HCV-14-2 B'FLY	Outside Inside	RC Pump Cooling Water Return	BYPASS/ TYPE C
25	Fuel Transfer Tube	Double Gasket Flange	Inside	Fuel Transfer	BYPASS/ TYPE C
26	CVCS	I-V-2516 Globe I-V-2522 Globe	Inside Outside	Letdown Line	BYPASS/ TYPE C

TABLE 3.6-1 (Continued)

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number/Type	Location to Containment	Service	Test Type*
28A	Sampling	ISE-05-1A,1B,1C,1D Globe ISE-05-1E Globe	Inside Outside	Safety/Injection Tank Sample	BYPASS TYPE C
28B	Sampling	I-V-5200 Globe I-V-5203 Globe	Inside Outside	RCS Hot Leg Sample	BYPASS TYPE C
29A	Sampling	I-V-5204 Globe I-V-5201 Globe	Outside Inside	Pressure Surge Sample	BYPASS TYPE C
29B	Sampling	I-V-5205 Globe I-V-5202 Globe	Outside Inside	Pressure Steam Sample	BYPASS TYPE C
31	Waste Management	I-V-6718 Diaph I-V-6750 Diaph	Inside Outside	Containment Vent Header	BYPASS TYPE C
41	Safety Injection	I-SE-03-2A,2B Globe I-V-3463 Gate	Inside Outside	Safety Injection Tank Fill/Drain and Sampling	BYPASS TYPE C
42	Waste Management	I-LCV-07-11A Globe I-LCV-07-11B Globe	Inside Outside	Reactor Cavity Sump Pump Discharge	BYPASS TYPE C
43	Waste Management	I-V-6341 Diaph I-V-6342 Diaph	Inside Outside	Reactor Drain Tank Pump Suction	BYPASS TYPE C
44	CVCS	I-V-2524 Globe I-V-2505 Globe	Inside Outside	Reactor Coolant Pump Controlled Bleedoff	BYPASS TYPE C
46	Fuel Pool	I-V-07-206 Gate I-V-07-189 Gate	Outside Inside	Fuel Pool Cleanup (inlet)	BYPASS TYPE C
47	Fuel Pool	I-V-07-170 Gate I-V-07-188 Gate	Outside Inside	Fuel Pool Cleanup (outlet)	BYPASS TYPE C

TABLE 3.6-1 (Continued)

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number/Type	Location to Containment	Service	Test Type*
48A	Sampling	I-FSE-27-8,9,10-11 Globe I-FSE-27-15 Globe	Inside Outside	H_2 Sampling Outlet	TYPE C
48B	Sampling	I-V-27-101 Check I-FSE-27-16 Globe	Inside Outside	H_2 Sampling Inlet	TYPE C
51A	Sampling	I-FSE-27-12,13,14 Globe I-FSE-27-18 Globe	Inside Outside	H_2 Sampling Outlet	TYPE C
51B	Sampling	I-V-27-102 Check I-FSE-27-17 Globe	Inside Outside	H_2 Sampling Inlet	TYPE C
52A	Sampling	I-FCV-26-1 Globe I-FCV-26-2 Globe	Inside Outside	Containment Radiation Monitoring	BYPASS TYPE C
52B	Sampling	I-FCV-26-3 Globe I-FCV-26-4 Globe	Inside Outside	Containment Radiation Monitoring	BYPASS TYPE C
52C	Sampling	I-FCV-26-5 Globe I-FCV-26-6 Globe	Inside Outside	Containment Radiation Monitoring	BYPASS TYPE C
52D	ILRT	I-V-00-140 Globe I-V-00-143 Globe	Inside Outside	ILRT	BYPASS TYPE C
52E	ILRT	I-V-00-139 Globe I-V-00-144 Globe	Inside Outside	ILRT ILRT	BYPASS TYPE C
54	ILRT	I-V-00-101 Gate Blind Flange	Outside Inside	ILRT	BYPASS/ TYPE C
56	Containment H_2 Purge	I-FCV-25-26 B'FLY I-V-25-25 Check	Outside Inside	Cont. Containment H_2 Purge Makeup Inlet	BYPASS/ TYPE C

TABLE 3.6-1 (Continued)

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number/Type	Location to Containment	Service	Test Type*
57	Containment H_2 Purge	I-V-FCV-25-20 B'FLY I-FCV-25-21 B'FLY	Inside Outside	Cont. Containment Purge - Exhaust	BYPASS/ TYPE C
67	Vacuum Relief	I-V-25-20 Check I-FCV-25-7 B'FLY	Inside Outside	Containment Vacuum Relief	TYPE C
68	Vacuum Relief	I-V-25-21 Check I-FCV-25-8 B'FLY	Inside Outside	Containment Vacuum Relief	TYPE C
Personnel Lock	NA	None	NA	Ingress and Egress To Containment	TYPE B
Escape Lock	NA	None	NA	Emergency Ingress and Egress to Containment	TYPE B
Maintenance Hatch	NA	None	NA	Vessel Maintenance	TYPE B
Electrical Penetrations	NA	All Primary Canisters except welded spares	NA		TYPE B
1	Main Steam Steel Containment Nozzle	Tap 1 Tap 2	Outside Outside	Expansion Bellows	TYPE B
2	Main Steam Steel Containment Nozzles	Tap 1 Tap 2	Outside Outside	Expansion Bellows	TYPE B
3	Feedwater Steel Containment Nozzles	Tap 1 Tap 2	Outside Outside	Expansion Bellows	TYPE B
4	Feedwater Steel Containment Nozzles	Tap 1 Tap 2	Outside Outside	Expansion Bellows	TYPE B
25	Fuel Tube Steel Containment Nozzles	Tap 1	Inside	Expansion Bellows	TYPE B
50	Temporary Services	Blind Flange Blind Flange	Outside Inside	Construction/Outage Use	TYPE B

*Type C and bypass tests are conducted in the same manner, the only difference is in the acceptance criteria that is applicable.

CONTAINMENT SYSTEMS

CONTAINMENT AIR LOCKS

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each containment air lock shall be OPERABLE with:

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

b. An overall air lock leakage rate of less than or equal to 0.05 L_a at P_a, 43.4 psig.

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

ACTION:

a. With one containment air lock door inoperable*:

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

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

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

4. The provisions of Specification 3.0.4 are not applicable.

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

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*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. Within 72 hours following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying the seal leakage is < 0.01 L_a as determined by precision flow measurement when the volume between the door seals is pressurized to greater than or equal to:

   1. For the personnel air lock, greater than or equal to P_a, 43.4 psig for at least 15 minutes if not tested with the automatic tester.

   2. For the emergency air lock, greater than or equal to 43.4 psig for at least 15 minutes.

b. By conducting overall air lock leakage tests at not less than P_a, 43.4 psig, and verifying the overall air lock leakage rate is within its limit:

   1. At least once per 6 months,# and

   2. Prior to establishing CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

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

#The provisions of Specification 4.0.2 are not applicable.

*This constitutes an exemption to Appendix J of 10 CFR 50.

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 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 during the shutdown for each Type A containment leakage rate test (reference Surveillance Requirement 4.6.1.2) 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. Any abnormal degradation of the containment vessel detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1.

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 less than or equal to 1000 hours per calendar year.

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 more than 1000 hours per calendar year, 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 STANDBY 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 The cumulative time that the 8-inch purge supply or exhaust isolation valves are open during the past calendar year shall be determined at least once per 7 days.

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 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 prior to entering MODE 4 from COLD SHUTDOWN if not tested within the previous 31 days.

CONTAINMENT SYSTEMS

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

CONTAINMENT SPRAY SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is positioned to take suction from the RWT 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 Specification 4.0.5.

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

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a 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 on OPERABLE shutdown cooling heat exchanger is established.

*Applicable only when pressurizer pressure is $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS (Continued)

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

d. At least once per 5 years by performing an air or smoke flow test through each spray header and verifying each spray nozzle is unobstructed.

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 $\geq$ 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 Specification 4.0.5.

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 $\geq$ 1750 psia.

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 Four independent containment fan coolers shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one of the above required containment fan coolers inoperable, restore the inoperable containment fan cooler to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours; restore the inoperable containment fan cooler to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment fan cooler shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Starting each fan cooler from the control room and verifying that each fan cooler operates for at least 15 minutes.

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

b. At least once per 18 months by verifying that each fan cooler starts automatically on an SIAS test signal.

CONTAINMENT SYSTEMS

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3 The containment isolation valves specified in Table 3.6-2 shall be OPERABLE with isolation times as shown in Table 3.6-2.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-2 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 isolation valves specified in Table 3.6-2 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 isolation valve specified in Table 3.6-2 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 valve of Table 3.6-2 shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Maximum Isolation Time (Sec)
A) Containment Isolation				
I-HCV-15-1	7	Primary Makeup Water (CIS)	Yes	5
I-HCV-18-1	9	Instrument Air Supply (CIS)	No	5
I-FCV-25-5,4	10	Containment Purge Exhaust (CIS)	No	3
I-FCV-25-2,3	11	Containment Purge Makeup (CIS)	No	3
V-6741	14	Nitrogen Supply to Safety Injection Tanks (CIS)	Yes	5
I-HCV-14-7 I-HCV-14-1	23	Reactor Coolant Pump Cooling Water Supply (SIAS)	No	5
I-HCV-14-6 I-HCV-14-2	24	Reactor Coolant Pump Cooling Water Return (SIAS)	No	5
I-HCV-2516 I-HCV-2522	26	Letdown Line (CIS)	No	5
I-SE-05-1A,1B, 1C,1D,1E	28A	Safety Injection Tank Sample	Yes	5
I-V-5200 I-V-5203	28B	Reactor Coolant System Hot Leg Sample (CIS)	Yes	5
I-V-5204 I-V-5201	29A	Pressurizer Surge Sample (CIS)	Yes	5
I-V-5205 I-V-5202	29B	Pressurizer Steam Sample (CIS)	Yes	5

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Maximum Isolation Time (Sec)
I-V-6718 I-V-6750	31	Containment Vent Header (CIS)	Yes	5
I-SE-03-2A,2B	41	Safety Injection Tank Test Line (CIS/SIAS)	Yes	5
I-LCV-07-11A I-LCV-07-11B	42	Reactor Cavity Sump Pump Discharge (CIS/SIAS)	Yes	5
I-V-6341 I-V-6342	43	RCDT Pump Suction (CIS)	Yes	5
I-V-2524 I-V-2505	44	RCP Controlled Bleed-off (CIS)	No	5
I-FCV-26-1 I-FCV-26-2	52A	Containment Radiation Monitoring (CIS)	Yes	10
I-FCV-26-3 I-FCV-26-4	52B	Containment Radiation Monitoring (CIS)	Yes	10
I-FCV-26-5 I-FCV-26-6	52C	Containment Radiation Monitoring (CIS)	Yes	10
I-FCV-25-26	56	Cont. Containment/H_2 Purge Makeup Inlet (CIS)	Yes	3
I-FCV-25-20 I-FCV-25-21	57	Cont. Containment/H_2 Purge Exhaust (CIS)	Yes	3

TABLE 3.6-2 (Continued)

CONTAINMENT ISOLATION VALVES

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Maximum Isolation Time (Sec)
B) Manual CR Remote Manual				
I-V-18-794 I-V-18-796 I-V-18-797	8	Station Air Supply (Manual)	Yes	NA
I-V-3463	41	Safety Injection Tank Test Line (Manual)	Yes	NA
I-V-07-206 I-V-07-189	46	Fuel Pool Cleanup (Inlet) (Manual)	Yes	NA
I-V-07-170 I-V-07-188	47	Fuel Pool Cleanup (Outlet) (Manual)	Yes	NA
I-FSE-27-8,9,10, 11,15,16	48	H_2 Sampling (Remote Manual)	Yes	NA
I-FSE-27-12,13,14, 17,18	51	H_2 Sampling (Remote Manual)	Yes	NA
I-V-00-140 I-V-00-143	52D	ILRT (Manual)	Yes	NA
I-V-00-139 I-V-00-144	52E	ILRT (Manual)	Yes	NA
I-V-00-101	54	ILRT (Manual)	Yes	NA

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 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. One volume percent hydrogen, balance nitrogen and oxygen.

b. Four volume percent hydrogen, balance nitrogen and oxygen.

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.

CONTAINMENT SYSTEMS

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5 The primary containment vessel to annulus vacuum relief valves shall be OPERABLE with an actuation set point of less than or equal to 9.85 ± 0.35 inches water gauge.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one primary containment vessel to annulus vacuum relief valve inoperable, restore the valve to OPERABLE status within 4 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

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: MODES 1, 2, 3, 4, 5, and 6 with operations involving movement of irradiated fuel within the spent fuel storage pool or crane operations with loads over the spent fuel storage pool with irradiated fuel in the spent fuel storage pool.

ACTION:

MODES 1, 2, 3, and 4:

With one SBVS 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.

ACTION:

MODES 5 and 6:

With one SBVS inoperable, restore the inoperable system to OPERABLE status within the next 7 days or suspend fuel movement within the spent fuel storage pool and crane operations over 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 iodine and ≥ 99% for radioactive elemental iodine when tested in accordance with ANSI N-510-1980 (130°C, 95% R.H.).

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 during the shutdown for each Type A containment leakage rate test (reference Surveillance Requirement 4.6.1.2) 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. Any abnormal degradation of the shield building detected during the above required inspections shall be reported to the Commission pursuant to Specification 6.9.1.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE with lift settings as specified in Table 3.7-1.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With both reactor coolant loops and associated steam generators in operation and with one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-2; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER			LIFT SETTING (±1%)*	RATED CAPACITY**
	Line No. 1	Line No. 2		
a.	8201	8205	1000 psia	744,210 lb/hr
b.	8202	8206	1000 psia	744,210 lb/hr
c.	8203	8207	1000 psia	744,210 lb/hr
d.	8204	8208	1000 psia	744,210 lb/hr
e.	8209	8213	1040 psia	774,000 lb/hr
f.	8210	8214	1040 psia	774,000 lb/hr
g.	8211	8215	1040 psia	774,000 lb/hr
h.	8212	8216	1040 psia	774,000 lb/hr

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

**Capacity is rated at lift setting +3% accumulation.

TABLE 3.7-2

MAXIMUM ALLOWABLE LINEAR 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 Linear Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
0	107.0
1	96.0
2	82.0
3	68.0
4	55.0

AUXILIARY FEEDWATER SYSTEM

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.

*The Auxiliary Feedwater System automatic initiation system shall be completely installed and OPERABLE prior to initial criticality.

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.

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 and in COLD SHUTDOWN within the following 24 hours.

MODES 2, 3 and 4 - With one main steam line isolation valve inoperable, subseqent operation in MODES 2, 3 or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 5.6 seconds on any closure actuation signal while in HOT STANDBY with $T_{avg} \geq 515°F$ during each reactor shutdown except that verification of full closure within 5.6 seconds need not be determined more often than once per 92 days.

MAIN FEEDWATER LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.6 Each main feedwater line isolation valve shall be OPERABLE.

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

ACTION:

MODE 1 - With one main feedwater 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 and in COLD SHUTDOWN within the following 24 hours.

MODES 2, 3 and 4 - With one main feedwater line isolation valve inoperable, subseqent operation in MODE 2, 3, or 4 may proceed provided:

a. The isolation valve is maintained closed.

b. The provisions of Specification 3.0.4 are not applicable.

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.6 Each main feedwater line isolation valve shall be demonstrated OPERABLE by:

a. Part-stroke exercising the valve at least once per 92 days, and

b. Verifying full closure within 5.6 seconds on any closure actuation signal while in HOT STANDBY with $T_{avg} \geq 515°F$ during each reactor shutdown except that verification of full closure within 5.35 seconds need not be determined more often than once per 92 days.

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 temperatures of both the primary and secondary coolants in the steam generators shall be greater than 100°F when the pressure of either coolant in the steam generator is greater than 275 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to less than or equal to 275 psig within 30 minutes, and

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

SURVEILLANCE REQUIREMENTS

4.7.2 The pressure in each side of the steam generators shall be determined to be less than 275 psig at least once per hour when the temperature of either the primary or secondary coolant is less than 100°F.

PLANT SYSTEMS

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

*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.

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 either a Hurricane Watch or a Hurricane Warning issued for the facility site, perform the St. Lucie Plant Beach Survey Procedure pursuant to Surveillance Requirement 4.7.6.1.1 below and 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

4.7.6.1.1 The St. Lucie Plant Beach Survey Procedure shall be conducted at least once per year between the dates of May 25 and June 7 and within 30 days following the termination of either a Hurricane Watch or a Hurricane Warning for the facility site. A Special Report containing the results of these surveys shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days following the completion of the survey. The Special Report shall include an evaluation of the facility flood protection if, as evidenced by this suryey program, the beach dune described in Specification 5.1.3 is lost.

4.7.6.1.2 The St. Lucie Mangrove Photographic Survey Procedure shall be conducted at least once per 12 months and shall be a color infrared photograph(s), or equivalent, of the mangrove area between the facility and the FP&L east property line. The results of these surveys shall be included in the Annual Operating Report for the period in which the survey was completed. This report shall include an evaluation of the facility flood protection if the survey indicates deterioration, either man-made or natural, of this mangrove area.

4.7.6.1.3 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.

PLANT SYSTEMS

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 in 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.

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 ± 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 ± 10%.

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

    5. Verifying a system flow rate of 2000 cfm ± 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 iodine when tested in accordance with ANSI N-510-1980 at 130°C, 95% R.H.

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 ± 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.

PLANT SYSTEMS

3/4.7.9 SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9 All snubbers listed in Tables 3.7-3a and 3.7-3b shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4. MODES 5 and 6 for snubbers located on systems required OPERABLE in those MODES.

ACTION:

With one or more snubbers inoperable, within 72 hours replace or restore the inoperable snubber(s) to OPERABLE status and perform an engineering evaluation per Specification 4.7.9g. 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

The first inservice visual inspection of each type of snubber shall be performed after 4 months but within 10 months of commencing POWER OPERATION and shall include all snubbers listed in Tables 3.7-3a and 3.7-3b. If less than two snubbers of any type are found inoperable during the first inservice visual inspection, the second inservice visual inspection shall be performed 12 months ± 25% from the date of the first inspection. Otherwise, subsequent visual inspections shall be performed in accordance with the following schedule:

No. Inoperable Snubbers of Each Type per Inspection Period	Subsequent Visual Inspection Period*#
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

*The inspection interval for each type of snubber shall not be lengthened more than one step at a time unless a generic problem has been identified and corrected; in that event the inspection interval may be lengthened one step the first time and two steps thereafter if no inoperable snubbers of that type are found.

#The provisions of Specification 4.0.2 are not applicable.

c. Refueling Outage Inspections

At least once per 18 months an inspection shall be performed of all snubbers listed in Tables 3.7-3a and 3.7-3b attached to sections of safety systems piping that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems. In addition to satisfying the visual inspection acceptance criteria, freedom of motion of mechanical snubbers shall be verified using one of the following: (1) manually induced snubber movement; (2) evaluation of in-place snubber piston setting; (3) stroking the mechanical snubber through its full range of travel.

d. Visual Inspection Acceptance Criteria

Visual inspections shall verify (1) that there are no visible indications of damage or impaired OPERABILITY, (2) attachments to the foundation or supporting structure are secure. Snubbers which appear inoperable as a result of visual inspections may be determined OPERABLE for the purpose of establishing the next visual inspection interval, provided that (1) the cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers, irrespective of type, that may be generically susceptible; and (2) the affected snubber is functionally tested in the as found condition and determined OPERABLE per Specification 4.7.9f. When a fluid port of a hydraulic snubber is found to be uncovered the snubber shall be declared inoperable and cannot be determined OPERABLE via functional testing unless the test is started with the piston in the as found setting, extending the piston rod in the tension mode direction. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be determined to be OPERABLE by visually verifying the required level of oil for operation for each affected snubber; otherwise declare the snubbers inoperable.

e. 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 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 4.7.9f. 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 4.7.9f. 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

SURVEILLANCE REQUIREMENTS (Continued)

Functional Tests (Continued)

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.

f. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range in both tension and compression.
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.
4. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement.
5. Fasteners for attachment of the snubber to the component and to the snubber anchorage are secure.

Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

g. Functional Test Failure Analysis

An engineering evaluation shall be made of each failure to meet the functional test acceptance criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of type which may be subject to the same failure mode.

Functional Test Failure Analysis (Continued)

For the snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components 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.

h. Functional Testing of Repaired and Replaced Snubbers

Snubbers which fail the visual inspection or the functional test acceptance criteria shall be repaired or replaced. Replacement snubbers and snubbers which have repairs which might affect the functional test result shall be tested to meet the functional test criteria before installation in the unit. These snubbers shall have met the acceptance criteria subsequent to their most recent service, and the functional test must have been performed within 12 months before being installed in the unit.

i. Snubber Seal Replacement Program

The seal service life of hydraulic snubbers shall be monitored to ensure that the seals 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.

j. 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. Snubbers so exempted shall continue to be listed in Tables 3.7-3a and 3.7-3b with footnotes indicating the extent of the exemptions.

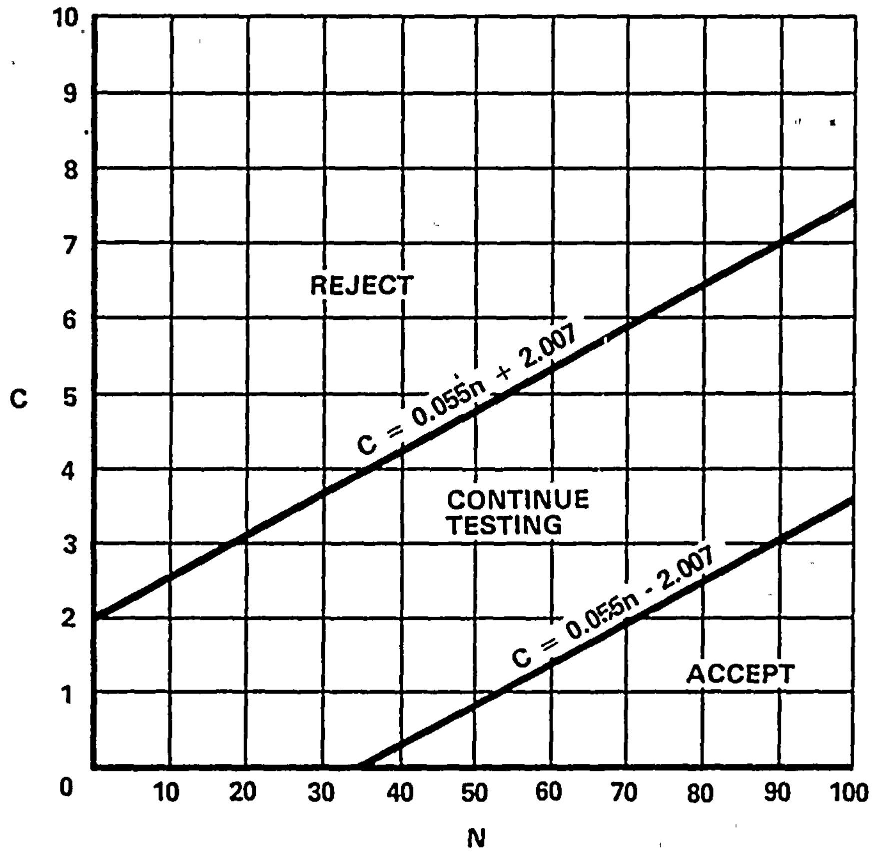


FIGURE 4.7-1 SAMPLING PLAN FOR SNUBBER FUNCTIONAL TEST

TABLE 3.7-3a

SAFETY-RELATED HYDRAULIC SNUBBERS*

Manufacturer: Paul Monroe

System	Quantity	Design Rating
R.C. Pumps	4	150.0 kips

Manufacturer: Taylor Devices

System	Quantity	Design Rating
Steam Generators**	16	440.0 kips

*Snubbers may be added or deleted to safety-related systems without prior License Amendment to Table 3.7-3a provided that a revision to Table 3.7-3a is included with the next License Amendment request.

**Steam generator snubbers are provided with an electrical switch for remote readout of fluid condition and are accessible for visual inspection during shutdown.

TABLE 3.7-3b

SAFETY-RELATED MECHANICAL SNUBBERS*

Manufacturer: Pacific Scientific

System	Small			Medium		Large	
	PSA-0.25**	PSA-0.5**	PSA-1.0**	PSA-3.0**	PSA-10.0**	PSA-35.0*	PSA-100.0*
Blowdown	4	1	0	0	0	0	0
Boiler Feedwater	0	0	2	9	16	5	0
Component Cooling	0	0	0	2	0	0	0
Chemical/Volume Cntl.	9	4	4	0	0	0	0
Containment Spray	0	0	2	0	0	0	0
Main Steam	5	1	6	19	15	29	6
Main Steam Instrumtn.	14	0	0	0	0	0	0
Pressurizer Spray	0	2	1	2	0	0	0
Reactor Coolant	30	0	11	13	10	0	0
Safety Injection	20	3	17	20	16	3	0
Waste Management	2	0	1	0	0	0	0
Type Total	84	11	44	65	57	37	6
Size Total			139		122		43
Total							294

*Snubbers may be added or deleted to safety-related systems without prior License Amendment to Table 3.7-3b provided that a revision to Table 3.7-3b is included with the next License Amendment request.

** Snubber Type: PSA-0.25 Design Load: 350 lbs

Snubber Type	Design Load
PSA-0.25	350 lbs
-0.5	650
-1.0	1,500
-3.0	6,000
-10.0	15,000
-35.0	50,000
-100.0	120,000

PLANT SYSTEMS

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.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

FIRE SUPPRESSION WATER SYSTEM*

LIMITING CONDITION FOR OPERATION

3.7.11.1 The fire suppression water system shall be OPERABLE with:

a. Two fire suppression pumps, each with a capacity of 2350 gpm, with their discharge aligned to the fire suppression header,

b. Separate water supplies, each with a minimum contained volume of 300,000 gallons, and

c. An OPERABLE flow path capable of taking suction from the city water storage tank 1A and the city water storage tank 1B and transferring the water through distribution piping with OPERABLE sectionalizing control or isolation valves to the yard hydrants, the last valve ahead of the water flow alarm device on each sprinkler or hose standpipe required to be OPERABLE per Specifications 3.7.11.2, 3.7.11.4 and 3.7.11.5.

APPLICABILITY: At all times.

ACTION:

a. With one pump and/or one water supply inoperable, restore the inoperable equipment to OPERABLE status within 7 days or, in lieu of any other report required by Specification 6.9.1, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within the next 30 days outlining the plans and procedures to be used to restore the inoperable equipment to OPERABLE status or to provide an alternate backup pump or supply. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

b. With the fire suppression water system otherwise inoperable, establish a backup fire suppression water system within 24 hours.

*This system is shared between St. Lucie Units 1 and 2.

SURVEILLANCE REQUIREMENTS

4.7.11.1 The fire suppression water system shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying the contained water supply volume.

b. At least once per 31 days on a STAGGERED TEST BASIS by starting each electric motor driven pump and operating it for at least 15 minutes on recirculation flow.

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

d. At least once per 12 months by performance of a system flush.

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

f. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. Verifying that each automatic valve in the flow path actuates to its correct position,

   2. Verifying that each pump develops at least 2350 gpm at a system head of 232 feet,

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

   4. Verifying that each fire suppression pump starts (sequentially) to maintain the fire suppression water system pressure greater than or equal to 85 psig.

g. At least once per 3 years by performing a flow test of the system in accordance with Chapter 5, Section 11 of the Fire Protection Handbook, 14th Edition, published by the National Fire Protection Association.

SPRINKLER SYSTEMS

LIMITING CONDITION FOR OPERATION

3.7.11.2 The following sprinkler systems shall be OPERABLE:

1. Fire Zone 8 - Diesel Generator Building 2A
2. Fire Zone 9 - Diesel Generator Building 2B
3. Fire Zone 19 - RAB East Hallway and Miscellaneous Equipment Areas*
4. Fire Zone 20 - RAB East-West Common Hallway*
5. Fire Zone 22 - RAB Electrical Penetration Area*
6. Fire Zone 23 - RAB Electrical Penetration Area*
7. Fire Zone 34 - Zone I - RAB Electrical Equipment Room*
8. Fire Zone 39 - RAB HVAC Equipment Room*
9. Fire Zone 51 - RAB Ceiling and Hallways*
10. Fire Zone 52 - Cable Spreading Room*

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

ACTION:

a. With one or more of the above required sprinkler systems inoperable, within 1 hour establish a continuous fire watch with backup fire suppression equipment for those areas in which redundant systems or components could be damaged; for other areas, establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.11.2 Each of the above required sprinkler systems shall be demonstrated OPERABLE:

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

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

*The sprinkler systems shall be completely installed and OPERABLE prior to exceeding 5% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months:

1. By performing a system functional test which includes simulated automatic actuation of the system, and:

    a) Verifying that the automatic valves in the flow path actuate to their correct positions on a thermal test signal, and

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

2. By a visual inspection of each sprinkler system to verify its integrity, and

3. By a visual inspection of each sprinkler's spray area to verify the spray pattern is not obstructed.

FIRE HOSE STATIONS

LIMITING CONDITION FOR OPERATION

3.7.11.3 The fire hose stations shown in Table 3.7-4 shall be OPERABLE.

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

ACTION:

a. With one or more of the fire hose stations shown in Table 3.7-4 inoperable, provide gated wye(s) on the nearest OPERABLE hose station(s). One outlet of the wye shall be connected to the standard length of hose provided for the hose station. The second outlet of the wye shall be connected to a length of hose sufficient to provide coverage for the area left unprotected by the inoperable hose station. Where it can be demonstrated that the physical routing of the fire hose would result in a recognizable hazard to operating technicians, plant equipment, or the hose itself, the fire hose shall be stored in a roll at the outlet of the operable hose station. Signs shall be mounted above the gated wye(s) to identify the proper hose to use. The above action shall be accomplished within 1 hour if the inoperable fire hose is the primary means of fire suppression; otherwise, route the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.11.3 Each of the fire hose stations shown in Table 3.7-4 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the stations accessible during plant operation to assure all required equipment is at the station.

b. For all fire hose stations located in the turbine building, at least once per 12 months by:

    1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

    2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

SURVEILLANCE REQUIREMENTS

c. At least once per 18 months by:

   1. Visual inspection of the stations not accessible during plant operations to assure all required equipment is at the station.

   2. Removing the hose for inspection and re-racking, and

   3. Inspecting all gaskets and replacing any degraded gaskets in the couplings.

d. For all fire hose stations not located in the turbine building, at least once per 3 years by:

   1. Partially opening each hose station valve to verify valve OPERABILITY and no flow blockage.

   2. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

TABLE 3.7-4

FIRE HOSE STATIONS

LOCATION/ELEVATION		HOSE RACK #
A.	Hose Stations (Turbine Building)	
1.	Operating Floor (northeast corner)	HS-15-4
2.	Operating Floor (southeast corner)	HS-15-10
3.	Operating Floor (middle east side)	HS-15-7
B.	Hose Stations (Reactor Auxiliary Building)	
1.	62 ft level east wall entrance	HS-15-44
2.	62 ft level west wall entrance	HS-15-45
3.	62 ft level west wall entrance to H&V room	HS-15-46
4.	43 ft level east wall (north of door)	HS-15-36
5.	43 ft level south wall of H&V room	HS-15-37
6.	43 ft level cable spreading room	HS-15-31
7.	43 ft level southwest corner of communications area near door	HS-15-42
8.	19.5 ft level east end of east-west hall	HS-15-38
9.	19.5 ft level middle of east-west hall	HS-15-40
10.	19.5 ft level south end of north-south hall	HS-15-33
11.	19.5 ft level entrance hall on south wall	HS-15-34
12.	-0.5 ft level east end of hall	HS-15-41
13.	-0.5 ft level south wall of hall	HS-15-28
14.	-0.5 ft level west end of hall	HS-15-43
C.	Hose Stations (Reactor Containment Building)	
1.	RCB at 23 ft level (near stairway no. 3)	HS-15-47
2.	RCB at 45 ft level (near stairway no. 1)	HS-15-48
3.	RCB at 45 ft level (near stairway no. 2)	HS-15-54
4.	RCB at 62 ft level (near stairway no. 3)	HS-15-49
D.	Hose Station (Fuel Handling Building)	
	62 ft level northwest corner	HS-15-55

YARD FIRE HYDRANTS AND HYDRANT HOSE HOUSES

LIMITING CONDITION FOR OPERATION

3.7.11.4 The yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be OPERABLE.

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

ACTION:

a. With one or more of the yard fire hydrants or associated hydrant hose houses shown in Table 3.7-5 inoperable, within 1 hour have sufficient additional lengths of 2 1/2 inch diameter hose located in an adjacent OPERABLE hydrant hose house to provide service to the unprotected area(s) if the inoperable fire hydrant or associated hydrant hose house is the primary means of fire suppression; otherwise provide the additional hose within 24 hours.

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

SURVEILLANCE REQUIREMENTS

4.7.11.4 Each of the yard fire hydrants and associated hydrant hose houses shown in Table 3.7-5 shall be demonstrated OPERABLE:

a. At least once per 31 days by visual inspection of the hydrant hose house to assure all required equipment is at the hose house.

b. At least once per 6 months (once during March, April or May and once during September, October or November) by visually inspecting each yard fire hydrant and verifying that the hydrant barrel is dry and that the hydrant is not damaged.

c. At least once per 12 months by:

   1. Conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above maximum fire main operating pressure, whichever is greater.

   2. Inspecting all the gaskets and replacing any degraded gaskets in the couplings.

   3. Performing a flow check of each hydrant to verify its OPERABILITY.

TABLE 3.7-5

YARD FIRE HYDRANTS AND ASSOCIATED HYDRANT HOSE HOUSES

LOCATION		HYDRANT NUMBER
RCB	NE	FH#6
RCB	NW	FH#7
CST	N	FH#9
CCWB	E	FH#20
DGB(2A)	E	FH#21
DGB(2B)	SE	FH#22
RAB	S	FH#23
Intake Structure	SE	FH#25
CST	SW	FH#26

PLANT SYSTEMS

3/4.7.12 FIRE RATED ASSEMBLIES

LIMITING CONDITION FOR OPERATION

3.7.12 All fire rated assemblies (walls, floor/ceilings, and other fire barriers) separating safety-related fire areas or separating portions of redundant systems important to safe shutdown within a fire area and all sealing devices in fire rated assembly penetrations (fire doors, fire dampers, cable, piping, and ventilation duct penetration seals) shall be OPERABLE.*

APPLICABILITY: At all times.

ACTION:

a. With one or more of the above required fire rated assemblies and/or sealing devices inoperable, within 1 hour either establish a continuous fire watch on at least one side of the affected assembly, or verify the OPERABILITY of the fire detectors on at least one side of the inoperable assembly and establish an hourly fire watch patrol.

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

SURVEILLANCE REQUIREMENTS

4.7.12.1 At least once per 18 months the above required fire rated assemblies and penetration sealing devices shall be verified OPERABLE by:

a. Performing a visual inspection of the exposed surfaces of each fire rated assembly.

b. Performing a visual inspection of each fire damper and associated hardware.

c. Performing a visual inspection of at least 10% of each type of sealed penetration. If apparent changes in appearance or abnormal degradations are found, a visual inspection of an additional 10% of each type of sealed penetration shall be made. This inspection process shall continue until a 10% sample with no apparent changes in appearance or abnormal degradation is found. Samples shall be selected such that each penetration seal will be inspected every 15 years.

*All fire rated assemblies shall be completely installed and OPERABLE prior to exceeding 5% of RATED THERMAL POWER, with the exception of the permanent flame impingement shields in containment which shall be completely installed and OPERABLE prior to STARTUP following the first refueling outage.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.12.2 Each of the above required fire doors shall be verified OPERABLE by inspecting the automatic hold-open, release and closing mechanism and latches at least once per 6 months, and by verifying:

a. That each unlocked fire door without electrical supervision is closed at least once per 24 hours.

b. That doors with automatic hold-open and release mechanisms are free of obstructions at least once per 24 hours.

c. That each locked closed fire door is closed at least once per 7 days.

4.7.12.3 At least once per 18 months, perform a functional test of each automatic hold-open and release mechanism.

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 either an offsite circuit other than the conditions delineated in Action 3.8.1.1f. or diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one offsite circuit and one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable sources to OPERABLE status within 12 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two offsite circuits and two diesel generators to OPERABLE status within 72 hours from the time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one diesel generator inoperable in addition to ACTION a. or b. above, verify that:

   1. All required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE, and

   2. When a MODE 1, 2, or 3, the steam-driven auxiliary feed pump is OPERABLE.

ACTION (Continued)

If these conditions are not satisfied within 2 hours be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. With two of the above required offsite A.C. circuits inoperable, demonstrate the OPERABILITY of two diesel generators by performing Surveillance Requirement 4.8.1.1.2a.4. within 1 hour and at least once per 8 hours thereafter, unless the diesel generators are already operating; restore at least one of the inoperable offsite sources to OPERABLE status within 24 hours or be in at least HOT STANDBY within the next 6 hours. With only one offsite source restored, restore at least two offsite circuits to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

e. With two of the above required diesel generators inoperable, demonstrate the OPERABILITY of two offsite A.C. circuits by performing Surveillance Requirement 4.8.1.1.1a. within 1 hour and at least once per 8 hours thereafter; restore at least one of the inoperable diesel generators to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore at least two diesel generators to OPERABLE status within 72 hours from time of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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 Requirements 4.8.1.1.1a. and 4.8.1.1.2a.4. 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 availablity, 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. In accordance with the frequency specified in Table 4.8-1 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 at least 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.

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 less than or equal to 60 seconds, and operates with a load greater than or equal to 3685 kW for at least an additional 60 minutes, and

6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the engine-mounted fuel tanks.

c. At least once per 92 days and from new fuel prior to addition to the storage tanks, by obtaining a sample of fuel oil in accordance with ASTM-D270-1975, and by verifying that the sample meets the following minimum requirements and is tested within the specified time limits:

1. As soon as sample is taken or prior to adding new fuel to the storage tank verify in accordance with the test specified in ASTM-D975-77 that the sample has:

SURVEILLANCE REQUIREMENTS (Continued)

a) A water and sediment content of less or equal to 0.05 volume percent.

b) A kinematic viscosity @ 40°C of greater than or equal to 1.9 centistokes, but less than or equal to 4.1 centistokes.

c) A specific gravity as specified by the manufacturer @ 60/60°F of greater than or equal to 0.8 but less than or equal to 0.99 or an API gravity @ 60°F of greater than or equal to 11 degrees but less than or equal to 47 degrees.

2. Within 1 week after obtaining the sample, verify an impurity level of less than 2 mg of insolubles per 100 ml when tested in accordance with ASTM-D2274-70.

3. Within 2 weeks of obtaining the sample verify that the other properties specified in Table 1 of ASTM-D975-77 and Regulatory Guide 1.137 Position 2.a are met when tested in accordance with ASTM-D975-77.

d. At least once per 12 months by verifying that the automatic load sequence timers are OPERABLE with the interval between each load block within ± 1 second of its design interval.

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

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

2. Verifying the generator capability to reject a load of greater than or equal to 453 kW while maintaining voltage at 4160 $\pm$ 420 volts and frequency at 60 $\pm$ 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.

   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 to greater than or equal to 3985 kW and during the remaining 22 hours of this test, the diesel generator shall be loaded to greater than or equal 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. Within 5 minutes after completing this 24-hour test, perform Surveillance Requirement 4.8.1.1.2e.4.b).

8. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000-hour rating of 3985 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 loads to the offsite power source, and

    c) Be restored to its standby status.

10. Verifying that with the diesel generator operating in a test mode (connected to its bus), a simulated safety injection signal overrides the test mode by (1) returning the diesel generator to standby operation and (2) automatically energizes the emergency loads with offsite power.

11. Verifying that the fuel transfer pump transfers fuel from each fuel storage tank to the engine-mounted tanks of each diesel via the installed cross connection lines.

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 at least 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 at a test pressure equal to 110% of the system design pressure.

SURVEILLANCE REQUIREMENTS (Continued)

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

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.

TABLE 4.8-1

DIESEL GENERATOR TEST SCHEDULE

Number of Failures In Last 100 Valid Tests.*	Test Frequency
$\leq$ 1	At least once per 31 days
2	At least once per 14 days
3	At least once per 7 days
$\geq$ 4	At least once per 3 days

*Criteria for determining number of failures and number of valid tests shall be in accordance with Regulatory Position C.2.e of Regulatory Guide 1.108, Revision 1, August 1977, where the last 100 tests are determined on a per nuclear unit basis. For the purposes of this test schedule, only valid tests conducted after the Operating License issuance date shall be included in the computation of the "last 100 valid tests". Entry into this test schedule shall be made at the 31 day test frequency.

ELECTRICAL POWER SYSTEMS

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 each 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, positive reactivity changes, 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 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1, 4.8.1.1.2 (except for requirement 4.8.1.1.2a.5) and 4.8.1.1.3.

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 ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	≥ 1.195(b)	≥ 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 ≥ 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, positive reactivity changes 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 bank by perfoming 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.

*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.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One train of A.C. emergency busses consisting of 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 electical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, 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.

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-3615-3645	LPSI INJ.	YES
	V-3616-3646	HPSI INJ.	YES
	V-3617-3647	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-V3644	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
	MV-21-4A,4B	ICW ISOL.	YES
CCW:	MV-14-17,18,19,20	FUEL POOL ISOL.	YES
	MV-14-9 TO 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 TO 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	H_2 CONT. PURGE	YES
	FCV-25-30,31	SFP EXHAUST	YES
	FCV-25-32,33	SBVS INLET	YES

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

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

a. Either a K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to 1720 ppm.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 40 gpm of a solution containing 1720 ppm boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to 1720 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position within the reactor pressure vessel.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

*The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the reactor vessel head closure bolts less than fully tensioned or with the head removed.

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 positive reactivity changes.

b. With both of the above required monitors inoperable or not operating, determine the boron concentration of the Reactor Coolant System at least once per 12 hours.

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.

REFUELING OPERATIONS

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, and

c. Each penetration providing direct access from the containment atmosphere to the outside atmosphere shall be either:

   1. Closed by an isolation valve, blind flange, or manual valve, or

   2. Be capable of being closed by an OPERABLE automatic containment isolation valve.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the containment.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or movement of irradiated fuel in the containment building.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment building penetrations shall be determined to be either in its closed/isolated condition or capable of being closed by an OPERABLE automatic containment isolation valve within 72 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS or movement of irradiated fuel in the containment building by:

a. Verifying the penetrations are in their closed/isolated condition, or

b. Testing the containment isolation valves per the applicable portions of Specification 4.6.3.2.

3/4.9.5 COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

3.9.5 Direct communications shall be maintained between the control room and personnel at the refueling station.*

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS.

SURVEILLANCE REQUIREMENTS

4.9.5 Direct communications between the control room and personnel at the refueling station shall be demonstrated within 1 hour prior to the start of and at least once per 12 hours during CORE ALTERATIONS.

*The sound powered telephone system shall be completely installed and OPERABLE prior to exceeding 5% of RATED THERMAL POWER.

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 when the crane load exceeds 3000 pounds.

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 the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System 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 one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

*The shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODE 6 when the water level above the top of the reactor pressure vessel flange is less than 23 feet.

ACTION:

a. With less than the required 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 all operations involving a reduction in boron concentration of the Reactor Coolant System 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 one shutdown cooling loop shall be verified to be in operation and circulating reactor coolant at a flow rate of greater than or equal to 3000 gpm at least once per 12 hours.

* Prior to initial criticality, the shutdown cooling loop may be removed from operation for up to 1 hour per 8-hour period during the performance of CORE ALTERATIONS in the vicinity of the reactor pressure vessel hot legs.

3/4.9.9 CONTAINMENT ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment isolation system shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS or movement of irradiated fuel within the 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. The provisions of Specification 3.0.4 are not applicable.

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 CORE ALTERATIONS 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 movement of fuel assemblies or CEAs within the reactor pressure vessel when either the fuel assemblies being moved or the fuel assemblies seated within the reactor pressure vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving movement of fuel assemblies or CEAs within the pressure vessel.

SURVEILLANCE REQUIREMENTS

4.9.10 The water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours thereafter during movement of fuel assemblies or CEAs.

3/4.9.11 WATER LEVEL-SPENT FUEL STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

APPLICABILITY: Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours.

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.

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 24 hours 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.

SPECIAL TEST EXCEPTIONS

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 Specifications 4.2.1.4 and 3.3.3.2 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.

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 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

3/4.10.5 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 3.3.3.2 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.

3/4.11 RADIOACTIVE EFFLUENTS

3/4.11.1 LIQUID EFFLUENTS

CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.11.1.1 The concentration of radioactive material released from the site (see Figure 5.1-1) shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 fpr radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to 2×10^{-4} microcuries/ml total activity.

APPLICABILITY: At all times.

ACTION:

With the concentration of radioactive material released from the site exceeding the above limits, immediately restore the concentration to within the above limits.

SURVEILLANCE REQUIREMENTS

4.11.1.1.1 The radioactivity content of each batch of radioactive liquid waste shall be determined prior to release by sampling and analysis in accordance with Table 4.11-1. The results of pre-release analyses shall be used with the calculational methods in the ODCM to assure that the concentration at the point of release is maintained within the limits of Specification 3.11.1.1. Also, results of the previous post-release analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release were maintained within the limits of Specification 3.11.1.1.

4.11.1.1.2 Post-release analyses of samples composited from batch releases shall be performed in accordance with Table 4.11-1.

4.11.1.1.3 The radioactivity concentration of liquids discharged from continuous release points shall be determined by collection and analysis of samples in accordance wiht Table 4.11-1. The results of the analyses shall be used with the calculational methods in the ODCM to assure that the concentrations at the point of release are maintained within the limits of Specification 3.11.1.1.

TABLE 4.11-1

RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM

Liquid Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD) (μCi/ml)[a]
A. Batch Waste Release Tanks[c]	P Each Batch	P Each Batch	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}
	P One Batch/M	M	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	P Each Batch	M Composite[b]	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	P Each Batch	Q Composite[b]	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
B. Continuous Releases[dg]	Daily	4/M Composite	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}
	D Grab Sample Daily	4/M Composite	Dissolved and Entrained Gases (Gamma Emitters)	1×10^{-5}
	Daily	M Composite	H-3	1×10^{-5}
			Gross Alpha	1×10^{-7}
	Daily	Q Composite	Sr-89, Sr-90	5×10^{-8}
			Fe-55	1×10^{-6}
C. Settling Basin[f]	W Grab Sample	W	Principal Gamma Emitters[e]	5×10^{-7}
			I-131	1×10^{-6}

TABLE 4.11-1 (Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.11-1 (Continued)

TABLE NOTATION

b. A composite sample is one in which the quantity of liquid samples is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen which is representative of the liquids released.

c. A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated, and then thoroughly mixed to assure representative sampling.

d. A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measurable and identifiable, together with the above nuclides, shall also be identified and reported.

f. Grab samples to be taken when there is confirmed primary to secondary system leakage indicated by the air ejector monitor indicating $\geq$ 2x background.

g. If Component Cooling Water activity is $>1\times10^{-5}$ µCi/ml, perform a weekly gross activity on the Intake Cooling Water System outlet to ensure the activity level is $\leq$ a 2×10^{-7} µCi/ml LLD limit. If ICW is $>2\times10^{-7}$ µCi/ml, perform analysis in accordance with a Plant Continuous Release on this Table.

RADIOACTIVE EFFLUENTS

DOSE

LIMITING CONDITION FOR OPERATION

3.11.1.2 The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each reactor unit, to UNRESTRICTED AREAS (see Figure 5.1-1) shall be limited:

a. During any calendar quarter to less than or equal to 1.5 mrems to the total body and to less than or equal to 5 mrems to any organ, and

b. During any calendar year to less than or equal to 3 mrems to the total body and to less than or equal to 10 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of radioactive materials in liquid effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases and radioactive materials in liquid effluents during the remainder of the current calendar quarter and during the subsequent three calendar quarters, so that the cumulative dose or dose commitment to an individual from these releases is within 3 mrems to the total body and 10 mrems to any organ.

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

SURVEILLANCE REQUIREMENTS

4.11.1.2 Dose Calculations. Cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

LIMITING CONDITION FOR OPERATION

3.11.1.3 The liquid radwaste treatment system shall be OPERABLE. The appropriate portions of the system shall be used to reduce the radioactive materials in liquid wastes prior to their discharge when the projected doses due to the liquid effluent from the site to UNRESTRICTED AREAS (see Figure 5.1-1) when averaged over 31 days, would exceed 0.06 mrem to the total body or 0.2 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the liquid radwaste treatment system inoperable for more than 31 days or with radioactive liquid waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days pursuant to Specification 6.9.2 a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.1.3.1 Doses due to liquid releases to UNRESTRICTED AREAS shall be projected at least once per 31 days, in accordance with the ODCM unless the liquid radwaste treatment system is being used.

4.11.1.3.2 The liquid radwaste treatment system shall be demonstrated OPERABLE by operating the liquid radwaste treatment system equipment for at least 30 minutes at least once per 92 days unless the liquid radwaste system has been utilized to process radioactive liquid effluents during the previous 92 days.

DOSE RATE

LIMITING CONDITION FOR OPERATION

3.11.2.1 The dose rate in UNRESTRICTED AREAS due to radioactive materials released in gaseous effluents from the site (see Figure 5.1-1) shall be limited to the following:

a. For noble gases: Less than or equal to 500 mrems/yr to the total body and less than or equal to 3000 mrems/yr to the skin, and

b. For iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days: Less than or equal to 1500 mrems/yr to any organ.

APPLICABILITY: At all times.

ACTION:

With the dose rate(s) exceeding the above limits, immediately decrease the release rate to within the above limit(s).

SURVEILLANCE REQUIREMENTS

4.11.2.1.1 The dose rate due to noble gases in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM.

4.11.2.1.2 The dose rate due to iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits in accordance with the methodology and parameters in the ODCM by obtaining representative samples and performing analyses in accordance with the sampling and analysis program specified in Table 4.11-2.

TABLE 4.11-2

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM

Gaseous Release Type	Sampling Frequency	Minimum Analysis Frequency	Type of Activity Analysis	Lower Limit of Detection (LLD)[a] (μCi/ml)
A. Waste Gas Storage Tank	P Each Tank Grab Sample	P Each Tank	Principal Gamma Emitters[e]	1×10^{-4}
B. Containment Purge	P Each Purge[b] Grab Sample	P Each Purge[b]	Principal Gamma Emitters[e]	1×10^{-4}
			H-3	1×10^{-6}
C. (1) Plant Vent (2) Fuel Building Vent (3) Laundry Area Vent (4) Steam Generator Blowdown Building Vent	4/M Grab Sample	4/M	Principal Gamma Emitters[e]	1×10^{-4}
			H-3	1×10^{-6}
D. All Release Types as listed in A, B, C above.	Continuous[d]	4/M[c] Charcoal Sample	I-131	1×10^{-12}
	Continuous[d]	4/M[c] Particulate Sample	Principal Gamma Emitters[e] (I-131, Others)	1×10^{-11}
	Continuous[d]	M Composite Particulate Sample	Gross Alpha	1×10^{-11}
	Continuous[d]	Q Composite Particulate Sample	Sr-89, Sr-90	1×10^{-11}
	Continuous[d]	Noble Gas Monitor	Noble Gases Gross Beta or Gamma	1×10^{-6}

TABLE 4.11-2 (Continued)

TABLE NOTATION

a. The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.11-2 (Continued)

TABLE NOTATION

b. Sampling and analysis shall also be performed following shutdown, startup, or a THERMAL POWER change exceeding 15% of RATED THERMAL POWER within 1 hour unless (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has not increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has not increased by more than a factor of 3.

c. Samples shall be changed at least 4 times a month and analyses shall be completed within 48 hours after changing (or after removal from sampler). Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup or THERMAL POWER change exceeding 15% of RATED THERMAL POWER in 1 hour and analyses shall be completed within 48 hours of changing if (1) analysis shows that the DOSE EQUIVALENT I-131 concentration in the primary coolant has increased more than a factor of 3; and (2) the noble gas activity monitor shows that effluent activity has increased by more than a factor or 3. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10.

d. The ratio of the sample flow rate to the sampled stream flow rate shall be known for the time period covered by each dose or dose rate calculation made in accordance with Specifications 3.11.2.1, 3.11.2.2 and 3.11.2.3.

e. The principal gamma emitters for which the LLD specification applies exclusively are the following radionuclides: Kr-87, Kr-88, Xe-133, Xe-133m, Xe-135, and Xe-138 for gaseous emissions and Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144 for particulate emissions. This list does not mean that only these nuclides are to be detected and reported. Other peaks which are measureable and identifiable, together with the above nuclides, shall also be identified and reported.

RADIOACTIVE EFFLUENTS

DOSE - NOBLE GASES

LIMITING CONDITION FOR OPERATION

3.11.2.2 The air dose due to noble gases released in gaseous effluents, from each reactor unit, to areas at and beyond the SITE BOUNDARY (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 5 mrads for gamma radiation and less than or equal to 10 mrads for beta radiation and,

b. During any calendar year: Less than or equal to 10 mrads for gamma radiation and less than or equal to 20 mrads for beta radiation.

APPLICABILITY: At all times.

ACTION

a. With the calculated air dose from radioactive noble gases in gaseous effluents exceeding any of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.2 Dose Calculations. Cumulative dose contributions for the current calendar quarter and current calendar year shall be determined in accordance with the methodolgy and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

LIMITING CONDITION FOR OPERATION

3.11.2.3 The dose to a MEMBER OF THE PUBLIC from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each reactor unit to areas at and beyond the SITE BOUNDARY, (see Figure 5.1-1) shall be limited to the following:

a. During any calendar quarter: Less than or equal to 7.5 mrems to any organ and,

b. During any calendar year: Less than or equal to 15 mrems to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the calculated dose from the release of iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit and defines the corrective actions that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits.

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

SURVEILLANCE REQUIREMENTS

4.11.2.3 Cumulative dose contributions for the current calendar quarter and current calendar year for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in the ODCM at least once per 31 days.

RADIOACTIVE EFFLUENTS

GASEOUS RADWASTE TREATMENT

LIMITING CONDITION FOR OPERATION

3.11.2.4 The GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM shall be OPERABLE. The appropriate portions of the GASEOUS RADWASTE TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected gaseous effluent air doses due to gaseous effluent releases from the site to UNRESTRICTED AREAS (see Figure 5.1-1), when averaged over 31 days, would exceed 0.2 mrad for gamma radiation and 0.4 mrad for beta radiation. The appropriate portions of the VENTILATION EXHAUST TREATMENT SYSTEM shall be used to reduce radioactive materials in gaseous waste prior to their discharge when the projected doses due to gaseous effluent releases from the site (see Figure 5.1-1) when averaged over 31 days would exceed 0.3 mrem to any organ.

APPLICABILITY: At all times.

ACTION:

a. With the GASEOUS RADWASTE TREATMENT SYSTEM and/or the VENTILATION EXHAUST TREATMENT SYSTEM inoperable for more than 31 days or with gaseous waste being discharged without treatment and in excess of the above limits, in lieu of any other report required by Specification 6.9.1, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report which includes the following information:

   1. Identification of the inoperable equipment or subsystems and the reason for the inoperability,

   2. Action(s) taken to restore the inoperable equipment to OPERABLE status, and

   3. Summary description of action(s) taken to prevent a recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.2.4.1 Doses due to gaseous releases from the site to UNRESTRICTED AREAS shall be projected at least once per 31 days, in accordance with the ODCM unless the GASEOUS RADWASTE TREATMENT SYSTEM is being used.

4.11.2.4.2 The GASEOUS RADWASTE TREATMENT SYSTEM and VENTILATION EXHAUST TREATMENT SYSTEM shall be demonstrated OPERABLE by operating the GASEOUS RADWASTE TREATMENT SYSTEM equipment and VENTILATION EXHAUST TREATMENT SYSTEM equipment for at least 30 minutes, at least once per 92 days unless the appropriate system has been utilized to process radioactive gaseous effluents during the previous 92 days.

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 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 The concentrations of hydrogen and 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 with the hydrogen and oxygen monitors required OPERABLE by Table 3.3-13 of Specification 3.3.3.10.

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, and provide report to the Commission pursuant to Specification 6.9.1.8.

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 $\frac{100}{\bar{E}}$.

3/4.11.3 SOLID RADIOACTIVE WASTE

LIMITING CONDITION FOR OPERATION

3.11.3 Radioactive bead resins shall be dewatered, as appropriate, in accordance with the PROCESS CONTROL PROGRAM to meet shipping and transportation requirements during transit, and disposal site requirements when received at the disposal site.

APPLICABILITY: At all times.

ACTION:

a. With dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately dewatered bead resin and correct the PROCESS CONTROL PROGRAM, the procedures and/or the dewatering system as necessary to prevent recurrence.

b. With dewatering not performed in accordance with the PROCESS CONTROL PROGRAM, (1) if the dewatered bead resin has not already been shipped for disposal, verify each container to ensure that it meets burial ground and shipping requirements and (2) take appropriate administrative action to prevent recurrence.

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

SURVEILLANCE REQUIREMENTS

4.11.3 Prior to disposal, each container of radioactive bead resins shall be tested for free standing liquids in accordance with the PROCESS CONTROL PROGRAM to assure that it meets shipping, transportation, and disposal site requirements.

3/4.11.4 TOTAL DOSE

LIMITING CONDITION FOR OPERATION

3.11.4 The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems.

APPLICABILITY: At all times.

ACTION:

a. With the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeding twice the limits of Specification 3.11.1.2a., 3.11.1.2b., 3.11.2.2a., 3.11.2.2b., 3.11.2.3a., or 3.11.2.3b., calculations shall be made including direct radiation contributions from the reactor units and from outside storage tanks to determine whether the above limits of Specification 3.11.4 have been exceeded. If such is the case, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that defines the corrective action to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and includes the schedule for achieving conformance with the above limits. This Special Report, as defined in 10 CFR Part 20.405c, shall include an analysis that estimates the radiation exposure (dose) to a MEMBER OF THE PUBLIC from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report.. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits, and if the release condition resulting in violation of 40 CFR Part 190 has not already been corrected, the Special Report shall include a request for a variance in accordance with the provisions of 40 CFR Part 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete.

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

SURVEILLANCE REQUIREMENTS

4.11.4.1 Cumulative dose contributions from liquid and gaseous effluents shall be determined in accordance with Surveillance Requirements 4.11.1.2, 4.11.2.2, and 4.11.2.3, and in accordance with the methodology and parameters in the ODCM.

4.11.4.2 Cumulative dose contributions from direct radiation from the reactor units and from radwaste storage tanks shall be determined in accordance with the methodology and parameters in the ODCM. This Surveillance Requirement shall be required only in the event the above Action a. requires the applicable calculations.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.1 MONITORING PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.1 The radiological environmental monitoring program shall be conducted as specified in Table 3.12-1.

APPLICABILITY: At all times.

ACTION:

a. With the radiological environmental monitoring program not being conducted as specified in Table 3.12-1, in lieu of a Licensee Event Report, prepare and submit to the Commission, in the Annual Radiological Environmental Operating Report required by Specification 6.9.1.11, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

b. With the confirmed* level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeding the reporting levels of Table 3.12-2 when averaged over any calendar quarter, in lieu of a Licensee Event Report, prepare and submit to the Commission within 30 days, pursuant to Specification 6.9.2, a Special Report that identifies the cause(s) for exceeding the limit(s) and defines the corrective actions to be taken to reduce radioactive effluents so that the potential annual dose to A MEMBER OF THE PUBLIC is less than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2, and 3.11.2.3. When more than one of the radionuclides in Table 3.12-2 are detected in the sampling medium, this report shall be submitted if:

$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \ldots \geq 1.0$$

When radionuclides other than those in Table 3.12-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose to A MEMBER OF THE PUBLIC is equal to or greater than the calendar year limits of Specifications 3.11.1.2, 3.11.2.2 and 3.11.2.3. This report shall include the methodology for calculating the cumulative potential dose contributions for the calendar year from radionuclides detected in environmental samples and can be determined in accordance with the methodology and parameters in the ODCM. This report is not required if the measured level of radioactivity was not the result of plant effluents; however, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Operating Report.

c. With milk or broadleaf vegetation samples unavailable from one or more of the sample locations required by Table 3.12-1, identify locations for obtaining replacement samples and add them to the radiological environmental monitoring program within 30 days. The specific

*A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable, as appropriate. The results of the confirmatory analysis shall be completed at the earliest time consistent with the analysis but in any case within 30 days.

ACTION: (Continued)

locations from which samples were unavailable may then be deleted from the monitoring program. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.10, identify the cause of the unavailability of samples and identify the new location(s) for obtaining replacement samples in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.1 The radiological environmental monitoring samples shall be collected pursuant to Table 3.12-1 from the specific locations given in the table and figure(s) in the ODCM, and shall be analyzed pursuant to the requirements of Table 3.12-1 and the detection capabilities required by Table 4.12-1.

TABLE 3.12-1

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM[a)]

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[b) c)]	Sampling and Collection Frequency[d)]	Type and Frequency[d)] of Analysis
1. DIRECT RADIATION[e)]	27 Monitoring Locations	Continuous monitoring with sample collection quarterly[f)]	Gamma exposure rate - quarterly.
2. AIRBORNE			
Radioiodine and Particulates	5 Locations	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading	Radioiodine Filter: I-131 analysis weekly Particulate Filter: Gross beta radioactivity analysis ≥ 24 hours following a filter change[g)] Gamma isotopic[h)] analysis of composite[g)] (by location) quarterly
3. WATERBORNE			
a. Surface[h)]	1 Location[i)]	Weekly	Gamma isotopic[h)] & tritium analyses weekly
	1 Location[j)]	Monthly	Gamma isotopic[h)] & tritium analyses monthly
b. Sediment from shoreline	2 Locations	Semiannually	Gamma isotopic[h)] analysis semiannually

TABLE 3.12-1 (Continued)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM[a)]

Exposure Pathway and/or Sample	Number of Representative Samples and Sample Locations[b) c)]	Sampling and Collection Frequency[d)]	Type and Frequency[d)] of Analysis
4. INGESTION			
a. Fish and Invertebrates			
1. Crustacea	2 Locations	Semiannually	Gamma isotopic[h)] analyses semiannually
2. Fish	2 Locations	Semiannually	Gamma isotopic[h)] analyses semiannually
b. Food Products			
1. Broad leaf vegetation	3 Locations[l)]	Monthly when available	Gamma isotopic[h)] and I-131 analyses monthly

TABLE 3.12-1 (continued)

TABLE NOTATION

a. Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, malfunction of automatic sampling equipment or other legitimate reasons. If specimens are unobtainable due to sampling equipment malfunction corrective action shall be taken prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.11.1.

b. Specific parameters of distance and direction sector from the centerline of one reactor, and additional description where pertinent, shall be provided for each sample location in Table 3.12-1 in a Table and figure(s) in the ODCM.

c. At times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions made within 30 days in the radiological environmental monitoring program.

d. The following definition of frequencies shall apply to Table 3.12-1 only:

<u>Weekly</u> - Not less than once per calendar week. A maximum interval of 11 days is allowed between the collection of any two consecutive samples.

<u>Semi-Monthly</u> - Not less than 2 times per calendar month with an interval of not less than 7 days between sample collections. A maximum interval of 24 days is allowed between collection of any two consecutive samples.

<u>Monthly</u> - Not less than once per calendar month with an interval of not less than 10 days between sample collections.

<u>Quarterly</u> - Not less than once per calendar quarter.

<u>Semiannually</u> - One sample each between calendar dates (January 1 - June 30) and (July 1 - December 31). An interval of not less than 30 days will be provided between sample collections.

The frequency of analyses is to be consistent with the sample collection frequency.

e. One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously may be used in place of, or in addition to, integrating dosimeters. For purposes of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters.

TABLE 3.12-1 (continued)

TABLE NOTATION

f. Refers to normal collection frequency. More frequent sample collection is permitted when conditions warrant.

g. Airborne particulate sample filters are analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. In addition to the requirement for a gamma isotopic on a composite sample a gamma isotopic is also required for each sample having a gross beta radioactivity which is > 1.0 pCi/m^3 and which is also > 10 times that of the most recent control sample.

h. Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.

i. Atlantic Ocean, in the vicinity of the public beaches along the eastern shore of Hutchinson Island near the St. Lucie Plant (grab sample).

j. Atlantic Ocean, at a location beyond influence from plant effluents (grab sample).

k. Discharges from the St. Lucie Plant do not influence drinking water or ground water pathways.

l. Samples of broad leaf vegetation grown nearest each of two different offsite locations of highest predicted annual average ground level D/Q, and one sample of similar broad leaf vegetation at an available location 15-30 km distant in the least prevalent wind direction based upon historical data in the ODCM.

TABLE 3.12-2

REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES

Reporting Levels

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gases (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/ℓ)	Food Products (pCi/kg, wet)
H-3	30,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-Nb-95***	400				
I-131	2**	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-La-140***	200			300	

*Since no drinking water pathway exists, a value of 30,000 pCi/ℓ is used. For drinking water samples, a value of 20,000 pCi/ℓ is used. This is 40 CFR Part 141 value.

**Applies to drinking water.

***An equilibrium mixture of the parent and daughter isotopes which corresponds to the reporting value of the parent isotope.

TABLE 4.12-1

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS[a]

LOWER LIMIT OF DETECTION (LLD)[b,c]

Analysis	Water (pCi/ℓ)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg,wet)	Milk (pCi/ℓ)	Food Products (pCi/kg,wet)	Sediment (pCi/kg,dry)
gross beta	4	0.01				
H-3	3000*					
Mn-54	15		130			
Fe-59	30		260			
Co-58,60	15		130			
Zn-65	30		260			
Zr-Nb-95	15[e]					
I-131	1[d]	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-La-140	15[e]			15[e]		

*Since no drinking water pathway exists, a value of 3000 pCi/ℓ may be used. For drinking water samples, a value of 2000 pCi/ℓ is used.

TABLE 4.12-1 (Continued)

TABLE NOTATION

[a]This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

[b]Required detection capabilities for thermoluminescent dosimeters used for environmental measurements are given in Regulatory Guide 4.13.

[c]The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95% probability with only 5% probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66\ s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda\Delta t)}$$

Where:

LLD is the "a priori" lower limit of detection as defined above, as picocuries per unit mass or volume,

s_b is the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate, as counts per minute,

E is the counting efficiency, as counts per disintegration,

V is the sample size in units of mass or volume,

2.22 is the number of disintegrations per minute per picocurie,

Y is the fractional radiochemical yield, when applicable,

λ is the radioactive decay constant for the particular radionuclide, and

Δt for environmental samples is the elapsed time between sample collection, or end of the sample collection period, and time of counting

Typical values of E, V, Y, and Δt should be used in the calculation.

TABLE 4.12-1 (Continued)

TABLE NOTATION

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

[d]LLD for drinking water samples. If no drinking water pathway exists, the LLD of gamma isotopic analysis may be used.

[e]An equilibrium mixture of the parent and daughter isotopes which corresponds to 15 pCi/ℓ of the parent isotope.

3/4.12.2 LAND USE CENSUS

LIMITING CONDITION FOR OPERATION

3.12.2 A land use census shall be conducted and shall identify within a distance of 8 km (5 miles) the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence and the nearest garden* of greater than 50 m^2 (500 ft^2) producing broad leaf vegetation.

APPLICABILITY: At all times.

ACTION:

a. With a land use census identifying a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in Specification 4.11.2.3, in lieu of a Licensee Event Report, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report, pursuant to Specification 6.9.1.10.

b. With a land use census identifying a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20% greater than at a location from which samples are currently being obtained in accordance with Specification 3.12.1, add the new location(s) to the radiological environmental monitoring program within 30 days. The sampling location(s), excluding the control station location, having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may be deleted from this monitoring program after October 31 of the year in which this land use census was conducted. In lieu of a Licensee Event Report and pursuant to Specification 6.9.1.10, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report and also include in the report a revised figure(s) and table for the ODCM reflecting the new location(s).

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

SURVEILLANCE REQUIREMENTS

4.12.2 The land use census shall be conducted during the growing season at least once per 12 months using that information that will provide the best results, such as by a door-to-door survey, aerial survey, or by consulting local agriculture authorities. The results of the land use census shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

*Broad leaf vegetation sampling may be performed at the site boundary in each of two different direction sectors with the highest predicted D/Qs in lieu of the garden census. Specifications for broad leaf vegetation sampling in Table 3.12-1.4b shall be followed, including analysis of control samples.

RADIOLOGICAL ENVIRONMENTAL MONITORING

3/4.12.3 INTERLABORATORY COMPARISON PROGRAM

LIMITING CONDITION FOR OPERATION

3.12.3 Analyses shall be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program that has been approved by the Commission.*

APPLICABILITY: At all times.

ACTION:

a. With analyses not being performed as required above, report the corrective actions to the Commission in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

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

SURVEILLANCE REQUIREMENTS

4.12.3 A summary of the results obtained as part of the above required Interlaboratory Comparison Program shall be included in the Annual Radiological Environmental Operating Report pursuant to Specification 6.9.1.11.

*This condition is satisfied by participation in the Environmental Radioactivity Laboratory Intercomparison Studies Program conducted by the Environmental Protection Agency (EPA).

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The BASES contained in the succeeding pages summarize the reasons for the specifications of Sections 3.0 and 4.0 but in accordance with 10 CFR 50.36 are not a part of these Technical Specifications.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the measures to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of a specification. For example, Specification 3.6.2.1 requires two Containment Spray Systems to be OPERABLE and provides explicit ACTION requirements if one spray system is inoperable. Under the terms of Specification 3.0.3, if both of the required Containment Spray Systems are inoperable, within one hour measures must be initiated to place the unit in at least HOT STANDBY within the next 6 hours, in at least HOT SHUTDOWN within the following 6 hours, and in COLD SHUTDOWN in the subsequent 24 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this specification have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

4.0.1 This specification provides that surveillance activities necessary to insure the Limiting Conditions for Operation are met and will be performed during the OPERATIONAL MODES or other conditions for which the Limiting Conditions for Operation are applicable. Provisions for additional surveillance activities to be performed without regard to the applicable OPERATIONAL MODES or other conditions are provided in the individual Surveillance Requirements. Surveillance Requirements for Special Test Exceptions need only be performed when the Special Test Exception is being utilized as an exception to an individual specification.

4.0.2 The provisions of this specification provide allowable tolerances for performing surveillance activities beyond those specified in the nominal surveillance interval. These tolerances are necessary to provide operational flexibility because of scheduling and performance considerations. The phrase "at least" associated with a surveillance frequency does not negate this allowable tolerance value and permits the performance of more frequent surveillance activities.

The tolerance values, taken either individually or consecutively over 3 test intervals, are sufficiently restrictive to ensure that the reliability associated with the surveillance activity is not significantly degraded beyond that obtained from the nominal specified interval.

4.0.3 The provisions of this specification set forth the criteria for determination of compliance with the OPERABILITY requirements of the Limiting Conditions for Operation. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these Technical Specifications.

This specification includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. This clarification is provided to ensure consistency in surveillance intervals thoughout these Technical Specifications and to remove any ambiguities relative to the frequencies for performing the required inservice inspection and testing activities.

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. For example, the requirements of Specification 4.0.4 to perform surveillance activities prior to entry into an OPERATIONAL MODE or other specified applicability condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps to be tested up to one week after return to normal operation. And for example, the Technical Specification definition of OPERABLE does not grant a grace period before a device that is not capable of performing its specified function is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 5.0% delta k/k is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN requirement is based upon this limiting condition and is consistent with FSAR safety analysis assumptions. At earlier times in core life, the minimum SHUTDOWN MARGIN required for the most restrictive conditions is less than 5.0% $\Delta k/k$. With T_{avg} less than or equal to 200°F, the reactivity transients resulting from any postulated accident are minimal and a 2% delta k/k SHUTDOWN MARGIN provides adequate protection.

3/4.1.1.3 BORON DILUTION

A minimum flow rate of at least 3000 gpm provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 3000 gpm will circulate an equivalent Reactor Coolant System volume of 10,931 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration reductions will therefore be within the capability of operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analysis remain valid through each fuel cycle. The surveillance requirements for measurement of the MTC during each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurances that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 515°F. This limitation is required to ensure (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) charging pumps, (3) separate flow paths, (4) boric acid makeup pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from expected operating conditions of 2.0% delta k/k after xenon decay and cooldown to 200°F. The maximum expected boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires boric acid solution from the boric acid makeup tanks in the allowable concentrations and volumes of Specification 3.1.2.8 or 72,000 gallons of 1720 ppm - 2100 ppm borated water from the refueling water tank.

With the RCS temperature below 200°F one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity changes in the event the single injection system becomes inoperable.

The boron capability required below 200°F is based upon providing a 2% delta k/k SHUTDOWN MARGIN after xenon decay and cooldown from 200°F to 140°F. This condition requires either 4,150 gallons of 1720 ppm - 2100 ppm borated water from the refueling water tank or boric acid solution from the boric acid makeup tanks in accordance with the requirements of Specification 3.1.2.7.

The contained water volume limits includes allowance for water not available because of discharge line location and other physical characteristics.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

3/4.1.2.9 BORON DILUTION

The simultaneous use of the boronometer and RCS sampling at intervals dependent upon the MODE and the number of OPERABLE charging pumps to monitor the RCS boron concentration provides diverse and redundant indications of an inadvertent boron dilution. This will allow detection with sufficient time for termination of the boron dilution event before a complete loss of SHUTDOWN MARGIN occurs.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of CEA misalignments are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original design criteria are met.

The ACTION statements applicable to a stuck or untrippable CEA, to two or more inoperable CEAs and to a large misalignment (greater than or equal to 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (less than 15 inches) of the CEAs, there is (1) a small effect on the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, (2) a small effect on the available SHUTDOWN MARGIN, and (3) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with small misalignments of CEAs permits a 1-hour time interval during which attempts may be made to restore the CEA to within its alignment requirements. The 1-hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs, and (3) minimize the effects of xenon redistribution.

MOVABLE CONTROL ASSEMBLIES (Continued)

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. This distribution may, in turn, have a significant effect on (1) the available SHUTDOWN MARGIN, (2) the time-dependent long-term power distributions relative to those used in generating LCOs and LSSS setpoints, and (3) the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in (1) local burnup, (2) peaking factors, and (3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

The requirement to reduce power in certain time limits depending upon the previous F_r^T is to eliminate a potential nonconservatism for situations when a CEA has been declared inoperable. A worst-case analysis has shown that a DNBR SAFDL violation may occur during the second hour after the CEA misalignment if this requirement is not met. This potential exists for cases in which the predeviation power level is between 70% and 80% and the Axial Shape Index is negative. This potential DNBR SAFDL violation is eliminated by the additional power reductions indicated. These reductions will be necessary once the deviated CEA has been declared inoperable. This time allowed to continued operation at a reduced power level can be permitted for the following reasons:

1. The margin calculations which support the Technical Specifications are based on a steady-state radial peak of F_r^T = 1.60.

2. When the actual F_r^T = 1.50, significant additional margin exists.

3. This additional margin can be credited to offset the increase in F_r^T with time that can occur following a CEA misalignment.

4. This increase in F_r^T is caused by xenon redistribution.

5. The present analysis can support allowing a misalignment to exist for up to 30 minutes without correction, if the initial $F_r^T \leq$ 1.50.

MOVABLE CONTROL ASSEMBLIES (Continued)

Operability of at least two CEA position indicator channels is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCOs are satisfied.

The maximum CEA drop time restriction is consistent with the assumed CEA drop time used in the safety analyses. Measurement with T_{avg} greater than or equal to 515°F and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

The LSSS setpoints and the power distribution LCOs were generated based upon a core burnup which would be achieved with the core operating in an essentially unrodded configuration. Therefore, the CEA insertion limit specifications require that during MODES 1 and 2, the full length CEAs be nearly fully withdrawn. The amount of CEA insertion permitted by the Long Term Steady State Insertion Limits of Specification 3.1.3.6 will not have a significant effect upon the unrodded burnup assumption but will still provide sufficient reactivity control. The Power Dependent Insertion Limits of Specification 3.1.3.6 are provided to ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels; however, long-term operation at these insertion limits could have adverse effects on core power distribution during subsequent operation in an unrodded configuration.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provides adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: (1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, (2) the flux peaking augmentation factors are as shown in Figure 4.2-1, (3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and (4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for (1) flux peaking augmentation factors as shown in Figure 4.2-1, (2) a measurement-calculational uncertainty factor of 1.062, (3) an engineering uncertainty factor of 1.03, (4) an allowance of 1.01 for axial fuel densification and thermal expansion, and (5) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F_{xy}^T AND F_r^T AND AXIMUTHAL POWER TILT - T_q

The limitations on F_{xy}^T and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F_r^T and T_q are provided to ensure that the assumptions used in the analysis establishing the DNB Margin LCO, the Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_{xy}^T, F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid.

An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The requirement that the measured value of Tq be mutiplied by the calculated values of F_r and F_{xy} to determine F_r^T and F_{xy}^T is applicable only when F_r and F_{xy} are calculated with a non-full core power distribution analysis code. When monitoring a reactor core power distribution, F_r or F_{xy} with a full core power distribution analysis code the azimuthal tilt is explicitly accounted for as part of the radial power distribution used to calculate F_{xy} and F_r.

The Surveillance Requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}, F_r and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

To ensure that the effect of rod bow on DNBR is conservatively accounted for, a penalty will be imposed on the measured value of F_r^T. The penalty is burnup dependent and includes a 1% penalty on DNBR due to grid spacing. The penalties to be applied to the measured F_r^T appear in Table B 3/4.2-1.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and safety analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of $\geq$ 1.20 throughout each analyzed transient.

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18-month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12-hour basis.

TABLE B 3/4.2-1

PENALTY TO BE APPLIED TO F_r^T TO ACCOUNT
FOR ROD BOW EFFECTS ON DNBR

BURNUP OF BUNDLE (GWd/MTU)	DNBR PENALTY (%)	DNBR PENALTY WITH GRID SPACING PENALTY (%)	PENALTY MULTIPLIER TO BE APPLIED TO MEASURED F_r^T
0-10.0	0.5	1.5	1.013
10.0-20.0	1.0	2.0	1.017
20.0-30.0	2.0	3.0	1.026
30.0-40.0	3.5	4.5	1.038
40.0-50.0	5.5	6.5	1.055

[illegible]

3/4.3.1 and 3/4.3.2 REACTOR PROTECTIVE AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS INSTRUMENTATION

The OPERABILITY of the reactor protective and Engineered Safety Features Actuation Systems instrumentation and bypasses ensure that (1) the associated Engineered Safety Features Actuation action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof reaches its setpoint, (2) the specified coincidence logic is maintained, (3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and (4) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundancy, and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the safety analyses.

The Surveillance Requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping, or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either (1) in place, onsite, or offsite test measurements or (2) utilizing replacement sensors with certified response times.

The Safety Injection Actuation Signal (SIAS) provides direct actuation of the Containment Isolation Signal (CIS) to ensure containment isolation in the event of a small break LOCA.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that: (1) the radiation levels are continually measured in the areas served by the individual channels; and (2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded; and (3) sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," December 1980 and NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event and evaluate the response of those features important to safety. This capability is required to permit comparison of the measured response to that used in the design basis for the facility to determine if plant shutdown is required pursuant to Appendix A of 10 CFR Part 100. The instrumentation is consistent with the recommendations of Regulatory Guide 1.12, "Instrumentation for Earthquakes," April 1974.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data are available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public.

3/4.3.3.5 REMOTE SHUTDOWN SYSTEM INSTRUMENTATION

The OPERABILITY of the remote shutdown system instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of 10 CFR Part 50.

The OPERABILITY of the remote shutdown system instrumentation ensures that a fire will not preclude achieving safe shutdown. The remote shutdown system instrumentation, control circuits, and transfer switches are independent of areas where a fire could damage systems normally used to shut down the reactor. This capability is consistent with General Design Criterion 3 and Appendix R to 10 CFR Part 50.

3/4.3.3.6 ACCIDENT MONITORING INSTRUMENTATION

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG 0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

3/4.3.3.7 FIRE DETECTION INSTRUMENTATION

OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required in order to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety-related equipment and is an integral element in the overall facility fire protection program.

In the event that a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire patrols in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

3/4.3.3.8 LOOSE-PART DETECTION INSTRUMENTATION

The OPERABILITY of the loose-part detection instrumentation ensures that sufficient capability is available to detect loose metallic parts in the primary system and avoid or mitigate damage to primary system components. The allowable out-of-service times and surveillance requirements are consistent with the recommendations of Regulatory Guide 1.133, "Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors," May 1981.

3/4.3.3.9 RADIOACTIVE LIQUID EFFLUENT MONITORING INSTRUMENTATION

The radioactive liquid effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in liquid effluents during actual or potential releases of liquid effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.3.10 RADIOACTIVE GASESOUS EFFLUENT MONITORING INSTRUMENTATION

The radioactive gaseous effluent instrumentation is provided to monitor and control, as applicable, the releases of radioactive materials in gaseous effluents during actual or potential releases of gaseous effluents. The alarm/trip setpoints for these instruments shall be calculated in accordance with the methodology in the ODCM to ensure that the alarm/trip will occur prior to exceeding the limits of 10 CFR Part 20. This instrumentation also includes provisions for monitoring the concentrations of potentially explosive gas mixtures in the waste gas holdup system. The OPERABILITY and use of this instrumentation is consistent with the requirements of General Design Criteria 60, 63 and 64 of Appendix A to 10 CFR Part 50.

3/4.3.4 TURBINE OVERSPEED PROTECTION

This specification is provided to ensure that the turbine overspeed protection instrumentation and the turbine speed control valves are OPERABLE and will protect the turbine from excessive overspeed. Protection from turbine excessive overspeed is required since excessive overspeed of the turbine could generate potentially damaging missiles which could impact and damage safety related components, equipment or structures.

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.20 during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations, and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one reactor coolant pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a reactor coolant pump in MODES 4 and 5, with one or more RCS cold legs less than or equal to 280°F during cooldown and 320°F during heatup are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR Part 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by (1) sizing each PORV to mitigate the pressure transient of an inadvertent safety injection actuation in a water-solid RCS with pressurizer heaters energized and (2) restricting starting of the RCPs to when the secondary water temperature of each steam generator is less than 100°F above each of the RCS cold leg temperatures.

3/4.4.2 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 212,182 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Protection System provides a diverse means of protection against RCS overpressurization at low temperatures.

SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the system pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power-operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Vessel Code.

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.4.4 PORV BLOCK VALVES

The power-operated relief valves (PORVs) and steam bubble function to relieve RCS pressure during all design transients up to and including the design step load decrease with steam dump. Operation of the PORVs in conjunction with a reactor trip on a Pressurizer Pressure-High signal minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The opening of the PORVs fulfills no safety-related function and no credit is taken for their operation in the safety analysis for MODE 1, 2, or 3.

Each PORV has a remotely operated block valve to provide a positive shutoff capability should a relief valve become inoperable. Since it is impractical and undesirable to actually open the PORVs to demonstate their reclosing, it becomes necessary to verify OPERABILITY of the PORV block valves to ensure the capability to isolate a malfunctioning PORV. As the PORVs are pilot operated and require some system pressure to operate, it is impractical to test them with the block valve closed.

The PORVs are sized to provide low temperature overpressure protection. As the capacity required to perform this function is excessive for operation in MODE 1, 2 or 3, it is necessary that the operation of more than one PORV be precluded during these MODES. Thus, one block valve is required to be shut during MODES 1, 2 and 3.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion.

STEAM GENERATORS (Continued)

Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these limits, localized corrosion may likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator tube leakage between the primary coolant system and the secondary coolant system primary-to-secondary leakage = 1.0 gpm from both steam generators. Cracks having a primary-to-secondary leakage less than this limit during operation will have an adequate margin of safety to withstand the loads imposed during normal operation and by postulated accidents. Operating plants have demonstrated that primary-to-secondary leakage of 0.5 gpm per steam generator can readily be detected by radiation monitors of steam generator blowdown. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior the resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of

LEAKAGE DETECTION SYSTEMS (Continued)

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 gpm leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the St. Lucie site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Operation with specific activity levels exceeding 1.0 microcurie/gram DOSE EQUIVALENT I-131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 800 hours per year (approximately 10 percent of the unit's yearly operating time) since the activity levels allowed by Figure 3.4-1 increase the 2-hour thyroid dose at the site boundary by a factor of up to 20 following a postulated steam generator tube rupture. The reporting of cumulative operating time over 500 hours in any 6 month consecutive period with greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 will allow sufficient time for Commission evaluation of the circumstances prior to reaching the 800-hour limit.

SPECIFIC ACTIVITY (Continued)

The sample analysis for determining the gross specific activity and $\bar{E}$ can exclude the radioiodines because of the low primary coolant limit of 1 microcurie/gram DOSE EQUIVALENT I-131, and because, if the limit is exceeded, the radioiodine level is to be determined every 4 hours. If the gross specific activity level and radioiodine level in the primary coolant were at their limits, the radioiodine contribution would be approximately 1%. In a release of primary coolant with a typical mixture of radioactivity, the actual radioiodine contribution would probably be about 20%. The exclusion of radionuclides with half-lives less than 10 minutes from these determinations has been made for several reasons. The first consideration is the difficulty in identifying short-lived radionuclides in a sample that requires a significant time to collect, transport, and analyze. The second consideration is the predictable delay time between the postulated release of radioactivity from the primary coolant to its release to the environment and transport to the SITE BOUNDARY, which is relatable to at least 30 minutes decay time. The choice of 10 minutes for the half-life cutoff was made because of the nuclear characteristics of the typical primary coolant radioactivity. The radionuclides in the typical primary coolant have half-lives of less than 4 minutes or half-lives of greater than 14 minutes, which allows a distinction between the radionuclides above and below a half-life of 10 minutes. For these reasons the radionuclides that are excluded from consideration are expected to decay to very low levels before thay could be transported from the primary coolant to the SITE BOUNDARY under any accident condition.

Based upon the above considerations for excluding certain radionuclides from the sample analysis, the allowable time of 2 hours between sample taking and completing the initial analysis is based upon a typical time necessary to perform the sampling, transport the sample, and perform the analysis of about 90 minutes. The gross count should be made in a reproducible geometry of sample and counter having reproducible γ or β self-shielding properties. The counter should be reset to a reproducible efficiency versus energy. It is not necessary to identify specific nuclides.

The determination of the contributors to the $\bar{E}$ result should be based upon those energy peaks identifiable with a 95% confidence level. The radiochemical determination of nuclides should be based on multiple counting of the sample with typical counting basis following sampling of less than 1 hour, about 2 hours, about 1 day, about 1 week, and about 1 month.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup and cooldown limit curves Figures 3.4-2, 3.4-3, and 3.4-4 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 100°F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period indicated on Figure 3.4-2.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these test are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2, 3.4-3, and 3.4-4 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

Piece No.	Code No.	Material	Vessel Location	Drop Weight Results	Temperature of Charpy V-Notch RT (°F) NDT @ 50 ft - lb	Minimum Upper Shelf Cv energy for Longitudinal Direction Charpy (1) Ft - lb
122-102A	M-604-1	SA 533B Cl 1	Upper Shell Plate	0	+50	-
122-102B	M-604-2	SA 533B Cl 1	Upper Shell Plate	+10	+50	-
122-102C	M-604-3	SA 533B Cl 1	Upper Shell Plate	-10	+10	-
124-102B	M-605-1	SA 533B Cl 1	Intermediate Shell Plate	0	+30	105
124-102C	M-605-2	SA 533B Cl 1	Intermediate Shell Plate	-10	+10	113
124-102A	M-605-3	SA 533B Cl 1	Intermediate Shell Plate	-20	-10	113
142-102C	M-4116-1	SA 533B Cl 1	Lower Shell Plate	-30	+20	91
142-102B	M-4116-2	SA 533B Cl 1	Lower Shell Plate	-50	+20	105
142-102A	M-4116-3	SA 533B Cl 1	Lower Shell Plate	-40	+20	100
102-101	M-4110-1	SA 533B Cl 1	Closure Head	-30	-10	-
106-101	M-4101-1	SA 508 Cl 2	Closure Head Flange	0	0	-
128-101A	M-4102-1	SA 508 Cl 2	Inlet Nozzle	-20	-20	-
128-101D	M-4102-2	SA 508 Cl 2	Inlet Nozzle	-20	-20	-
128-101B	M-4102-3	SA 508 Cl 2	Inlet Nozzle	0	0	-
128-101C	M-4102-4	SA 508 Cl 2	Inlet Nozzle	-10	-10	-
128-301B	M-4103-1	SA 508 Cl 2	Outlet Nozzle	-20	-20	-
128-301A	M-4103-2	SA 508 Cl 2	Outlet Nozzle	-30	-30	-
126-101	M-602-1	SA 508 Cl 2	Vessel Flange	-30	-10	-
131-102A	M-4104-1	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-102D	M-4104-2	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-102B	M-4104-3	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-102C	M-4104-4	SA 508 Cl 1	Inlet Nozzle Safe End	-20	+20	-
131-101B	M-4105-1	SA 508 Cl 1	Outlet Nozzle Safe End	-10	0	-
131-101A	M-4105-2	SA 508 Cl 1	Outlet Nozzle Safe End	-10	0	-
152-101	M-4112-1	SA 533B Cl 1	Bottom Head Dome	-50	-50	-
154-102 (A to F)	M-4111-1	SA 533B Cl 1	Bottom Head Torus	-40	+40	-
104-102 (A to D)	M-4109-1	SA 533B Cl 1	Closure Head Torus	-60	-10	-

(1) Reported only for beltline region plates.

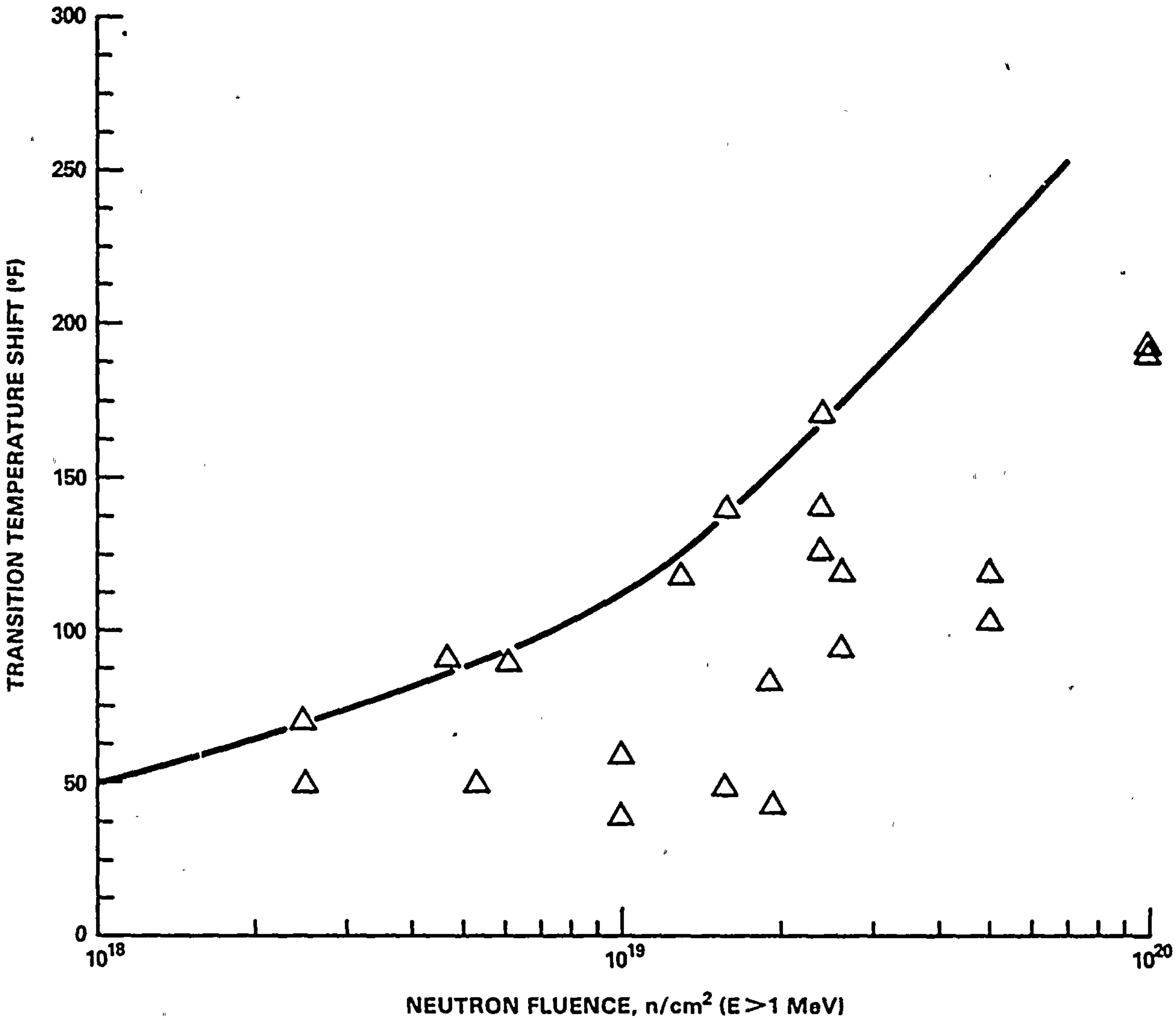


Figure B 3/4.4-1
Nil-ductility transition temperature increase as a function of fast (E>1 MeV) neutron fluence (550°F irradiation)

PRESSURE/TEMPERATURE LIMITS (Continued)

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the delta RT_{NDT} determined from the surveillance capsule is different from the calculated delta RTNDT for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figures 3.4-2, 3.4-3, and 3.4-4 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all Reactor Coolant System pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figures 3.4-2, 3.4-3, and 3.4-4 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F for piping, pumps, and valves. Below this temperature, the system pressure must be limited to a maximum of 20% of the system's hydrostatic test pressure of 3125 psia.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

The OPERABILITY of two PORVs or an RCS vent opening of greater than 3.58 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 280°F during cooldown and 320°F during heatup. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either (1) a safety injection actuation in a water-solid RCS with the pressurizer heaters energized or (2) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 100°F above the RCS cold leg temperatures with the pressurizer solid.

3/4.4.10 REACTOR COOLANT SYSTEM VENTS

Reactor Coolant System vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The OPERABILITY of at least one Reactor Coolant System vent path from the reactor vessel head and the pressurizer steam space ensures the capability exists to perform this function.

The redundancy design of the Reactor Coolant System vent systems serves to minimize the probability of inadvertent or irreversible actuation while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path.

The function, capabilities, and testing requirements of the Reactor Coolant System vent system are consistent with the requirements of Item II.b.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.

3/4.4.11 STRUCTURAL INTEGRITY

The inservice inspection and testing programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity and operational readiness of these components will be maintained at an acceptable level throughout the life of the plant. These programs are in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50.55a (g) (6) (i).

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1971 Edition and Addenda through Summer 1973.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Reactor Coolant System (RCS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the assumptions used for safety injection tank injection in the safety analysis are met.

The safety injection tank power-operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limits for operation with a safety injection tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occuring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one safety injection tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS hot leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

ECCS SUBSYSTEMS (Continued)

With the RCS temperature below 325°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The trisodium phosphate dodecahydrate (TSP) stored in dissolving baskets located in the containment basement is provided to minimize the possibility of corrosion cracking of certain metal components during operation of the ECCS following a LOCA. The TSP provided this protection by dissolving in the sump water and causing its final pH to be raised to greater than or equal to 7.0.

The limitation for a maximum of one high pressure safety injection pump to be OPERABLE, and the Surveillance Requirement to verify all high pressure safety injection pumps except the required OPERABLE pump to be inoperable below 280°F during cooldown and 320°F during heatup, although the analysis supports actuation of safety injection in a water solid RCS with pressurizer heaters energized, provide additional administrative assurance that a mass addition pressure transient can be relieved by the operation of a single PORV.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. Surveillance Requirements for throttle valve position stops and flow balance testing provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses. The requirement to dissolve a representative sample of TSP in a sample of RWT water provides assurance that the stored TSP will dissolve in borated water at the postulated post-LOCA temperatures.

3/4.5.4 REFUELING WATER TANK

The OPERABILITY of the Refueling Water Tank (RWT) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of a LOCA. The limits on RWT minimum volume and boron concentration ensure that (1) sufficient water is available within containment to permit recirculation cooling flow to the core, and (2) the reactor will remain subcritical in the cold condition following mixing of the RWT and the RCS water volumes with all control rods inserted except for the most reactive control assembly. These assumptions are consistent with the LOCA analyses.

REFUELING WATER TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The limits on contained water volume and boron concentration of the RWT also ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components.

BASES

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a or less than or equal to 0.75 L_t, as applicable during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix J of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate. Surveillance testing of the air lock seals provides assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that (1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.7 psi and (2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break conditions.

The maximum peak pressure expected to be obtained from a steam line break event is 43.4 psig. The limit of 0.4 psig for initial positive containment pressure will limit the total pressure to 43.99 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment temperature does not exceed the design temperature of 264°F during steam line break conditions and is consistent with the safety analyses.

3/4.6.1.6 CONTAINMENT VESSEL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment steel vessel will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the vessel will withstand the maximum pressure of 44.0 psig in the event of a steam line break accident. A visual inspection in conjunction with Type A leakage test is sufficient to demonstrate this capability.

3/4.6.1.7 CONTAINMENT VENTILATION SYSTEM

The 48-inch containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA or steam line break accident. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system. To provide assurance that the 48-inch valves cannot be inadvertently opened, they are sealed closed in accordance with Standard Review Plan 6.2.4 which includes devices to lock the valve closed, or prevent power from being supplied to the valve operator.

The use of the containment purge lines is restricted to the 8-inch purge supply and exhaust isolation valves since, unlike the 48-inch valves, the 8-inch valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization and cooling capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

CONTAINMENT SPRAY SYSTEM (Continued)

The Containment Spray System and the Containment Cooling System provide post-accident cooling of the containment atmosphere. However, the Containment Spray System also provides a mechanism for removing iodine from the containment atmosphere and therefore the time requirements for restoring an inoperable spray system to OPERABLE status have been maintained consistent with that assigned other inoperable ESF equipment.

3/4.6.2.2 IODINE REMOVAL SYSTEM

The OPERABILITY of the Iodine Removal System ensures that sufficient N_2H_4 is added to the containment spray in the event of a LOCA. The limits on N_2H_4 volume and concentration ensure a minimum of 50 ppm of N_2H_4 concentration available in the spray for a minimum of 6.5 hours per pump for a total of 13 hours to provide assumed iodine decontamination factors on the containment atmosphere during spray function and ensure a pH value of between 7.0 and 11.0 for the solution recirculated within containment after a LOCA. This pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the Containment Cooling System ensures that (1) the containment air temperature will be maintained within limits during normal operation, and (2) adequate heat removal capacity is available when operated in conjunction with the Containment Spray Systems during post-LOCA conditions.

The Containment Cooling System and the Containment Spray System provide post-accident cooling of the containment atmosphere. As a result of this cooling capability, the allowable out-of-service time requirements for the Containment Cooling System have been appropriately adjusted. The allowable out-of-service time requirements for the Containment Spray System and Containment Cooling System have been maintained consistent with that assigned other inoperable ESF equipment since the Containment Spray System and Containment Cooling System also provide a mechanism for removing iodine from the containment atmosphere.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through GDC 57 of Appendix A to 10 CFR Part 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

The containment fan coolers and containment spray ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

3/4.6.5 VACUUM RELIEF VALVES

The OPERABILITY of the primary containment to atmosphere vacuum relief valves ensures that the containment internal pressure differential does not become more negative than 0.615 psi. This condition is necessary to prevent exceeding the containment design limit for internal pressure differential of 0.7 psi.

3/4.6.6 SECONDARY CONTAINMENT

3/4.6.6.1 SHIELD BUILDING VENTILATION SYSTEM

The OPERABILITY of the shield building ventilation systems ensures that containment vessel leakage occurring during LOCA conditions into the annulus will be filtered through the HEPA filters and charcoal adsorber trains prior to discharge to the atmosphere and also reduces radioactive effluent releases to the environment during a fuel handling accident in the spent fuel storage building. This requirement is necessary to meet the assumptions used in the safety analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions.

Operation of the system with the heaters on for at least 10 hours continuous over a 31-day period is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

3/4.6.6.2 SHIELD BUILDING INTEGRITY

SHIELD BUILDING INTEGRITY ensures that the release of radioactive materials from the primary containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with operation of the shield building ventilation system, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.6.3 SHIELD BUILDING STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment shield building will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to provide (1) protection for the steel vessel from external missiles, (2) radiation shielding in the event of a LOCA, and (3) an annulus surrounding the steel vessel that can be maintained at a negative pressure during accident conditions. A visual inspection is sufficient to demonstrate this capability.

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1100 psia) of its design pressure of 1000 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 12,384,000 lbs/hr which is 110.0% of the total secondary steam flow of 11,172,000 lbs/hr at 100% RATED THERMAL POWER. A minimum of one OPERABLE safety valve per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \left(\frac{8-N}{8}\right) \times 110.0$$

where:

SP	=	reduced reactor trip setpoint in percent of RATED THERMAL POWER. This is a ratio of the available relieving capacity over the total steam flow at rated power.
8	=	total number of secondary safety valves for one steam generator.
N	=	The number of inoperable secondary safety valves on the steam generator with the greater number of inoperable valves.
110.0	=	the ratio of the total relieving capacity of all sixteen (16) secondary safety valves (12,384,000 lbs/hr at 1071 psia, maximum set pressure plus 3% accumulation) over the secondary steam flow at 100% Rated Thermal Load (11,172,000 lbs/hr).

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM

The OPERABILITY of the auxiliary feedwater system ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power.

Each electric-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 320 gpm at a pressure of 1000 psia to the entrance of the steam generators. The steam-driven auxiliary feedwater pump is capable of delivering a total feedwater flow of 500 gpm at a pressure of 1000 psia to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the shutdown cooling system may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available to maintain the Unit 2 RCS at HOT STANDBY conditions for 4 hours followed by an orderly cooldown to the shutdown cooling entry temperature (350°F). The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

The actual water requirements are 149,600 gallons for Unit 2 and 125,000 gallons for Unit 1. Included in the required volumes of water are the tank unusable volume of 9400 gallons and a conservative allowance for instrument error of 21,400 gallons.

3/4.7.1.4. ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose also includes the effects of a coincident 1.0 gpm primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements is consistent with the assumptions used in the safety analyses.

3/4.7.1.6 MAIN FEEDWATER LINE ISOLATION VALVES

The main feedwater line isolation valves are required to be OPERABLE to ensure that (1) feedwater is terminated to the affected steam generator following a steam line break and (2) auxiliary feedwater is delivered to the intact steam generator following a feedwater line break. If feedwater is not terminated to a steam generator with a broken main steam line, two serious effects may result: (1) the post-trip return to power due to plant cooldown will be greater with resultant higher fuel failure and (2) the steam released to containment will exceed the design.

Due to removal of the main feed check valve from the plant design and its replacement with a second main feedwater line isolation valve, there is nothing other than the main feedwater line isolation valves to prevent back flow of AFW following a feed line break. This may result in a loss of condensate inventory and the potential for not being able to feed the steam generator.

The concern is the failure of one main feedwater line isolation valve to close with the other main feedwater line isolation valve in that line being inoperable (i.e., stuck open). It is thus desired to preclude operation for extended periods with a main feedwater line isolation valve known to be stuck in the open position.

3/4.7.1.7 ATMOSPHERIC DUMP VALVES

The limitation on maintaining the atmospheric dump valves in the manual mode of operation is to ensure the atmospheric dump valves will be closed in the event of a steam line break. For the steam line break with atmospheric dump valve control failure event, the failure of the atmospheric dump valves to close would be a valid concern were the system to be in the automatic mode during power operations.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure-induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations to 100°F and 275 psig are based on a steam generator RT_{NDT} of 30°F and are sufficient to prevent brittle fracture.

3/4.7.3 COMPONENT COOLING WATER SYSTEM

The OPERABILITY of the Component Cooling Water System ensures that sufficient cooling capacity is available for continued operation of safety-related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 INTAKE COOLING WATER SYSTEM

The OPERABILITY of the Intake Cooling Water System ensures that sufficient cooling capacity is available for continued operation of equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level ensure that sufficient cooling capacity is available to either (1) provide normal cooldown of the facility, or (2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level is based on providing an adequate cooling water supply to safety-related equipment until cooling water can be supplied from Big Mud Creek.

Cooling capacity calculations are based on an ultimate heat sink temperature of 95°F. It has been demonstrated by a temperature survey conducted from March 1976 to May 1981 that the Atlantic Ocean has never risen higher than 86°F. Based on this conservatism, no ultimate heat sink temperature limitation is specified.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility protective actions will be taken in the event of flood conditions. The installation of the stoplogs ensures adequate protection for wave run-up effects where no permanent adjacent structures exist and provides protection to safety-related equipment. The maximum wave runup from the probable maximum flood (PMF) has been calculated to be elevation 18.0 feet Mean Low Water (MLW).

3/4.7.7 CONTROL ROOM EMERGENCY AIR CLEANUP SYSTEM

The OPERABILITY of the Control Room Emergency Air Cleanup System ensures that (1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and (2) the control room will remain habitable for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rems or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix A, 10 CFR Part 50.

3/4.7.8 ECCS AREA VENTILATION SYSTEM

The OPERABILITY of the ECCS Area Ventilation System ensures that cooling air is provided for ECCS equipment.

3/4.7.9 SNUBBERS

All snubbers are required OPERABLE to ensure that the structural integrity of the Reactor Coolant System and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads. Snubbers excluded from this inspection program are those installed on nonsafety related systems and then only if their failure or failure of the system on which they are installed, would have no adverse effect on any safety related system.

Snubbers are classified and grouped by design and manufacturer but not by size. For example, mechanical snubbers utilizing the same design features of the 2 kip, 10 kip and 100 kip capacity manufactured by company "A" are of the same type. The same design mechanical snubber manufactured by company "B", for purposes of this Specification, would be of a different type, as would hydraulic snubbers from either manufacturer.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

To provide assurance of snubber functional reliability, one of two sampling and acceptance criteria methods are used:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.7-1.

Figure 4.7-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

SNUBBERS (Continued)

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

Sealed sources are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism (i.e. sealed sources within radiation monitoring or boron measuring devices) are considered to be stored and need not be tested unless they are removed from the shield mechanism.

3/4.7.11 FIRE SUPPRESSION SYSTEMS

The OPERABILITY of the fire suppression systems ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety-related equipment is located. The fire suppression system consists of the water system, sprinklers, fire hose stations, and yard fire hydrants. The collective capability of the fire suppression systems is adequate to minimize potential damage to safety-related equipment and is a major element in the facility fire protection program.

In the event that portions of the fire suppression systems are inoperable, alternate backup fire fighting equipment is required to be made available in the affected areas until the inoperable equipment is restored to service. When the inoperable fire fighting equipment is intended for use as a backup means of fire suppression, a longer period of time is allowed to provide an alternate means of fire fighting than if the inoperable equipment is the primary means of fire suppression.

The Surveillance Requirements provide assurance that the minimum OPERABILITY requirements of the fire suppression systems are met.

FIRE SUPPRESSION SYSTEMS (Continued)

In the event the fire suppression water system becomes inoperable, immediate corrective measures must be taken since this system provides the major fire suppression capability of the plant.

3/4.7.12 FIRE RATED ASSEMBLIES

The OPERABILITY of the fire barriers and barrier penetrations ensures that fire damage will be limited. These design features minimize the possibility of a single fire involving more than one fire area prior to detection and extinguishment. The fire barriers, fire barrier penetrations for conduits, cable trays and piping, fire dampers, and fire doors are periodically inspected to verify their OPERABILITY.

3/4.8.1, 3/4.8.2 and 3/4.8.3 A.C. SOURCES, D.C SOURCES and ONSITE POWER DISTRIBUTION SYSTEMS

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one redundant set of onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source. The A.C. and D.C. source allowable out-of-service times are based on Regulatory Guide 1.93, "Availability of Electrical Power Sources," December 1974. When one diesel generator is inoperable, there is an additional ACTION requirement to verify that all required systems, subsystems, trains, components and devices, that depend on the remaining OPERABLE diesel generator as a source of emergency power, are also OPERABLE, and that the steam-driven auxiliary feedwater pump is OPERABLE. This requirement is intended to provide assurance that a loss of offsite power event will not result in a complete loss of safety function of critical systems during the period one of the diesel generators is inoperable. The term verify as used in this context means to administratively check by examining logs or other information to determine if certain components are out-of-service for maintenance or other reasons. It does not mean to perform the surveillance requirements needed to demonstrate the OPERABILITY of the component.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the unit status.

The Surveillance Requirements for demonstrating the OPERABILITY of the diesel generators are in accordance with the recommendations of Regulatory Guides 1.9 "Selection of Diesel Generator Set Capacity for Standby Power Supplies," March 10, 1971, and 1.108 "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants," Revision 1, August 1977, and 1.137, "Fuel Oil Systems for Standby Diesel Generators," Revision 1, October 1979.

A.C. SOURCES, D.C. SOURCES AND ONSITE POWER DISTRIBUTION SYSTEMS (Continued)

The Surveillance Requirement for demonstrating the OPERABILITY of the Station batteries are based on the recommendations of Regulatory Guide 1.129, "Maintenance Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants," February 1978, and IEEE Std 450-1980, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations."

Verifying average electrolyte temperature above the minimum for which the battery was sized, total battery terminal voltage on float charge, connection resistance values and the performance of battery service and discharge tests ensures the effectiveness of the charging system, the ability to handle high discharge rates and compares the battery capacity at that time with the rated capacity.

Table 4.8-2 specifies the normal limits for each designated pilot cell and each connected cell for electrolyte level, float voltage and specific gravity. The limits for the designated pilot cells float voltage and specific gravity, greater than 2.13 volts and .015 below the manufacturer's full charge specific gravity or a battery charger current that had stabilized at a low value, is characteristic of a charged cell with adequate capacity. The normal limits for each connected cell for float voltage and specific gravity, greater than 2.13 volts and not more than .020 below the manufacturer's full charge specific gravity with an average specific gravity of all the connected cells not more than .010 below the manufacturer's full charge specific gravity, ensures the OPERABILITY and capability of the battery.

Operation with a battery cell's parameter outside the normal limit but within the allowable value specified in Table 4.8-2 is permitted for up to 7 days. During this 7 day period: (1) the allowable values for electrolyte level ensures no physical damage to the plates with an adequate electron transfer capability; (2) the allowable value for the average specific gravity of all the cells, not more than .020 below the manufacturer's recommended full charge specific gravity, ensures that the decrease in rating will be less than the safety margin provided in sizing; (3) the allowable value for an individual cell's specific gravity, ensures that an individual cell's specific gravity will not be more than .040 below the manufacturer's full charge specific gravity and that the overall capability of the battery will be maintained within an acceptable limit; and (4) the allowable value for an individual cell's float voltage, greater than 2.07 volts, ensures the battery's capability to perform its design function.

3/4.8.4 ELECTRICAL EQUIPMENT PROTECTIVE DEVICES

The OPERABILITY of the motor operated valves thermal overload protection and/or bypass devices ensures that these devices will not prevent safety related valves from performing their function. The Surveillance Requirements for demonstrating the OPERABILITY of these devices are in accordance with Regulatory Guide 1.106 "Thermal Overload Protection for Electric Motors on Motor Operated Valves", Revision 1, March 1977.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: (1) the reactor will remain subcritical during CORE ALTERATIONS, and (2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the safety analyses. The value of 0.95 or less for K_{eff} includes a 1% delta k/k conservative allowance for uncertainties. Similarly, the boron concentration value of 1720 ppm or greater also includes a conservative uncertainty allowance of 50 ppm boron.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the startup neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety analyses.

3/4.9.4 CONTAINMENT BUILDING PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE

The OPERABILITY requirements for the refueling machine ensure that: (1) manipulator cranes will be used for movement of fuel assemblies, with or without CEAs, (2) each crane has sufficient load capacity to lift a fuel assembly, with or without CEAs, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly, CEA and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 SHUTDOWN COOLING AND COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.

The requirement to have two shutdown cooling loops OPERABLE when there is less than 23 feet of water above the reactor pressure vessel flange with irradiated fuel in the core ensures that a single failure of the operating shutdown cooling loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the reactor pressure vessel flange with irradiated fuel in the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating shutdown cooling loop, adequate time is provided to initiate emergency procedures to cool the core.

3/4.9.9 CONTAINMENT ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment isolation valves will be automatically isolated upon detection of high radiation levels within the containment. The OPERABILITY of this system is required to restrict the release of radioactive material from the containment atmosphere to the environment.

3/4.9.10 and 3/4.9.11 WATER LEVEL-REACTOR VESSEL and STORAGE POOL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 SPENT FUEL CASK CRANE

The maximum load which may be handled by the spent fuel cask crane is limited to a loaded multi-element cask which is equivalent to approximately 100 tons. This restriction is provided to ensure the structural integrity of the spent fuel pool in the event of a dropped cask accident. Structural damage caused by dropping a load in excess of a loaded multi-element cask could cause leakage from the spent fuel pool in excess of the maximum makeup capability.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when CEA worth measurement tests are performed. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

Although CEA worth testing is conducted in MODE 2, during the performance of these tests sufficient negative reactivity is inserted to result in temporary entry into MODE 3. Because the intent is to immediately return to MODE 2 to continue CEA worth measurements, the special test exception allows limited operation in MODE 3 without having to borate to meet the SHUTDOWN MARGIN requirements of Technical Specification 3.1.1.1.

3/4.10.2 MODERATOR TEMPERATURE COEFFICIENT, GROUP HEIGHT, INSERTION, AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to (1) measure CEA worth and (2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 REACTOR COOLANT LOOPS

This special test exception permits reactor criticality under reduced flow conditions and is required to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

3/4.10.5 CEA INSERTION DURING ITC, MTC, AND POWER COEFFICIENT MEASUREMENTS

This special test exception permits the CEA groups to be misaligned during such PHYSICS TESTS as those required to determine the (1) isothermal temperature coefficient, (2) moderator temperature coefficient, and (3) power coefficient.

3/4.11 RADIOACTIVE EFFLUENTS

BASES

3/4.11.1 LIQUID EFFLUENTS

3/4.11.1.1 CONCENTRATION

This specification is provided to ensure that the concentration of radioactive materials released in liquid waste effluents from the site to UNRESTRICTED AREAS will be less than the concentration levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2. This limitation provides additional assurance that the levels of radioactive materials in bodies of water outside the site will result in exposures within (1) the Section II.A design objectives of Appendix I, 10 CFR Part 50, to a individual, and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

3/4.11.1.2 DOSE

This specification is provided to implement the requirements of Sections II.A, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.A of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in liquid effluents will be kept "as low as is reasonably achievable." The dose calculation methodology and parameters in the ODCM implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The equations specified in the ODCM for calculating the doses due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," March, 1976 and Regulatory Guide 1.113, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

This specification applies to the release of liquid effluents from each reactor at the site. For units with shared radwaste treatment systems, the liquid effluents from the shared system are proportioned among the units sharing that system.

3/4.11.1.3 LIQUID WASTE TREATMENT

The OPERABILITY of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50 and the design objective given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives set forth in Section II.A of Appendix I, 10 CFR Part 50, for liquid effluents.

3/4.11.2 GASEOUS EFFLUENTS

3/4.11.2.1 DOSE RATE

This specification is provided to ensure that the dose at any time at and beyond the SITE BOUNDARY from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR Part 20 to UNRESTRICTED AREAS. The annual dose limits are the doses associated with the concentrations of 10 CFR Part 20, Appendix B, Table II, Column 1. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a MEMBER OF THE PUBLIC in an UNRESTRICTED AREA, either within or outside the SITE BOUNDARY, to annual average concentrations exceeding the limits specified in Appendix B, Table II of 10 CFR Part 20 (10 CFR Part 20.106(b)). For MEMBERS OF THE PUBLIC who may at times be within the SITE BOUNDARY, the occupancy of that MEMBER OF THE PUBLIC will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the SITE BOUNDARY. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a MEMBER OF THE PUBLIC at or beyond the SITE BOUNDARY to less than or equal to 500 mrems/year to the total body or to less than or equal to 3000 mrems/year to the skin. These release rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child via the inhalation pathway to less than or equal to 1500 mrems/year.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the lower limits of detection (LLDs). Detailed discussion of the LLD, and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215 (June 1975).

This specification applies to the release of gaseous effluents from all reactors at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.2 DOSE - NOBLE GASES

This specification is provided to implement the requirements of Sections II.B, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Condition for Operation implements the guides set forth in Section II.B of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," March, 1976 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the SITE BOUNDARY are based upon the historical average atmospheric conditions or real atmospheric conditions.

This specification applies to the release of gaseous effluents from each reactor at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.3 DOSE - IODINE-131, IODINE-133, TRITIUM, AND RADIONUCLIDES IN PARTICULATE FORM

This specification is provided to implement the requirements of Sections II.C, III.A and IV.A of Appendix I, 10 CFR Part 50. The Limiting Conditions for Operation are the guides set forth in Section II.C of Appendix I. The ACTION statements provide the required operating flexibility and at the same time implement the guides set forth in Section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to UNRESTRICTED AREAS will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the Surveillance Requirements implement the requirements in Section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a MEMBER OF THE PUBLIC through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," March, 1976 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric or real meteorological conditions. The release rate specifications for iodine-131, iodine-133, tritium, and radionuclides in particulate form with half lives greater than 8 days are dependent upon the existing radionuclide pathways to man, in the areas at and beyond the SITE BOUNDARY. The pathways that were examined in the development of these calculations were: (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat producing animals graze with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

This specification applies to the release of gaseous effluents from each reactor at the site. For units with shared radwaste treatment systems, the gaseous effluents from the shared system are proportioned among the units sharing that system.

3/4.11.2.4 GASEOUS RADWASTE TREATMENT

The OPERABILITY of the GASEOUS RADWASTE TREATMENT SYSTEM and the VENTILATION EXHAUST TREATMENT SYSTEM ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable". This specification implements the requirements of 10 CFR Part 50.36a, General Design Criterion 60 of Appendix A to 10 CFR Part 50, and the design objectives given in Section II.D of Appendix I to 10 CFR Part 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in Sections II.B and II.C of Appendix I, 10 CFR Part 50, for gaseous effluents.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 GAS STORAGE TANKS

Restricting the quantity of radioactivity contained in each gas storage tank provides assurance that in the event of an uncontrolled release of the tank's contents, the resulting total body exposure to an individual at the nearest exclusion area boundary will not exceed 0.5 rem. This is consistent with Standard Review Plan 15.7.1, "Waste Gas System Failure."

3/4.11.3 SOLID RADIOACTIVE WASTE

This specification implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

3/4.11.4 TOTAL DOSE

This specification is provided to meet the dose limitations of 40 CFR Part 190 that have been incorporated into 10 CFR Part 20 by 46 FR 18525. The specification requires the preparation and submittal of a Special Report whenever the calculated doses from plant generated radioactive effluents and direct radiation exceed 25 mrems to the total body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrems. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a MEMBER OF THE PUBLIC will exceed the dose limits of 40 CFR Part 190 if the individual reactors remain within twice the dose design objectives of Appendix I, and if direct radiation doses from the reactor units and outside storage tanks are kept small. The Special Report will describe a course of action that should result in the limitation of the annual dose to a MEMBER OF THE PUBLIC to within the 40 CFR Part 190 limits. For the purposes of the Special Report, it may be assumed that the dose commitment to the MEMBER OF THE PUBLIC from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any MEMBER OF THE PUBLIC is estimated to exceed the requirements of 40 CFR Part 190, the Special Report with a request for a variance (provided the release conditions resulting in violation of 40 CFR Part 190 have not already been corrected), in accordance with the provisions of 40 CFR Part 190.11 and 10 CFR Part 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR Part 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR Part 190, and does not apply in any way to the other requirements for dose limitation of 10 CFR Part 20, as addressed in Specifications 3.11.1.1 and 3.11.2.1. An individual is not considered a MEMBER OF THE PUBLIC during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

3/4.12 RADIOLOGICAL ENVIRONMENTAL MONITORING

BASES

3/4.12.1 MONITORING PROGRAM

The radiological environmental monitoring program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposures of MEMBERS OF THE PUBLIC resulting from the station operation. This monitoring program implements Section IV.B.2 of Appendix I to 10 CFR Part 50 and thereby supplements the radiological effluent monitoring program by verifying that the measurable concentrations of radioactive materials and levels of radiation are not higher than expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the lower limits of detection (LLDs). The LLDs required by Table 4.12-1 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an _a priori_ (before the fact) limit representing the capability of a measurement system and not as an _a posteriori_ (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in HASL Procedures Manual, _HASL-300_ (revised annually), Currie, L. A., "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," _Anal. Chem. 40_, 586-93 (1968), and Hartwell, J. K., "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report _ARH-SA-215_ (June 1975).

3/4.12.2 LAND USE CENSUS

This specification is provided to ensure that changes in the use of areas at and beyond the SITE BOUNDARY are identified and that modifications to the radiological environmental monitoring program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of Section IV.B.3 of Appendix I to 10 CFR Part 50. Restricting the census to gardens of greater than 50 m^2 provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity 26 kg/year of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: 1) 20% of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage), and 2) a vegetation yield of 2 kg/m^2.

3/4.12/3 INTERLABORATORY COMPARISON PROGRAM

The requirement for participation in an approved Interlaboratory Comparison Program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are reasonably valid for the purposes of Section IV.B.2 of Appendix I to 10 CFR Part 50.

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 = 2.5×10^6 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.

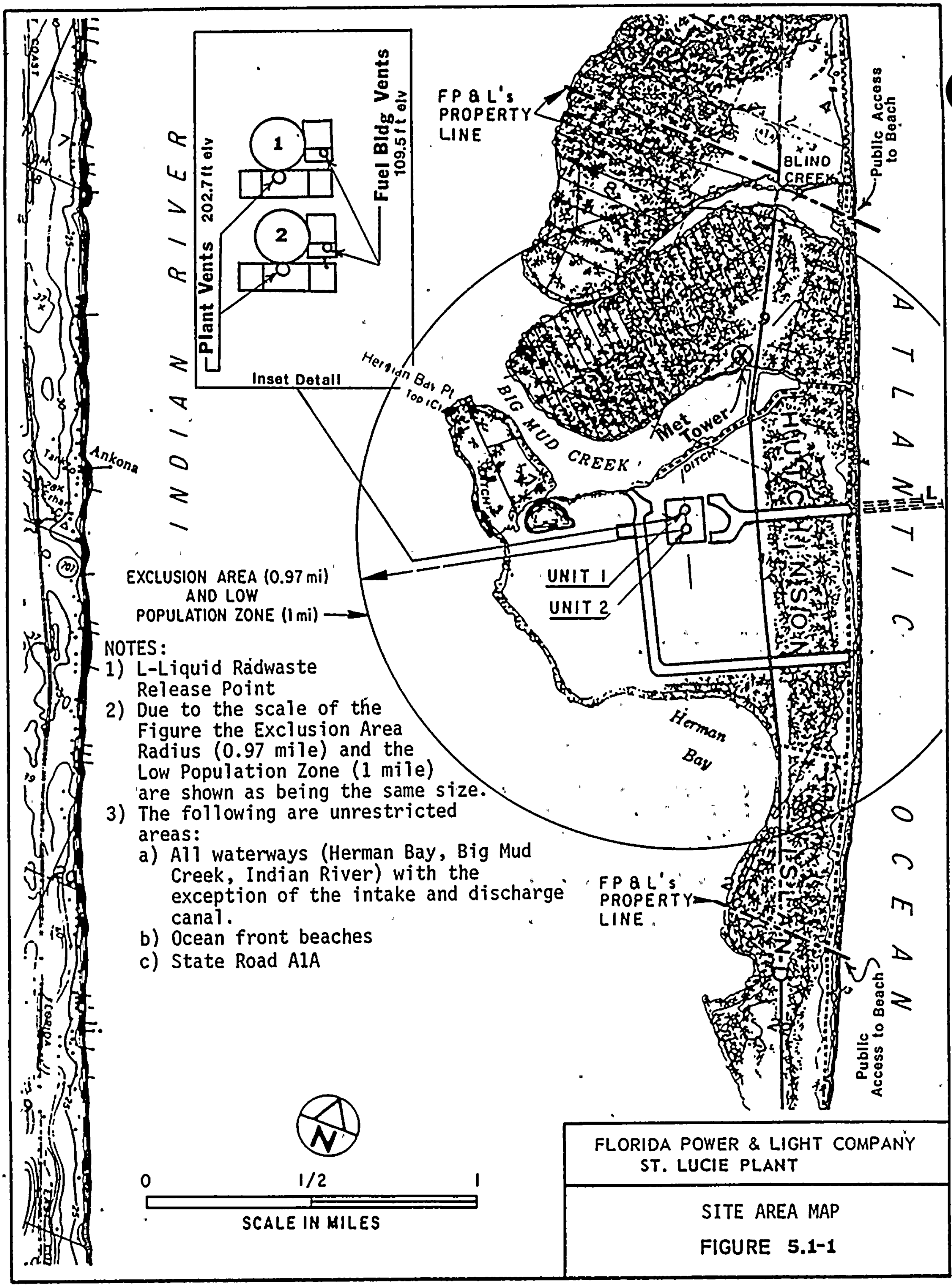
e. Cylinder wall minimum thickness = 3 feet.

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 rods clad with Zircaloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 1698.5 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 and shall have a maximum enrichment of 3.70 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 83 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 of 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 ± 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 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties of 0.021 k_{eff} and calculational margin of 0.079 k_{eff} as described in St. Lucie Unit 2 response to NRC Question 410.11 in Section 9.1.2 of the FSAR.

b. A nominal 14 inch center-to-center distance between fuel assemblies placed in the storage racks.

The k_{eff} for new fuel for the first core loading stored dry in the spent fuel storage racks shall not exceed 0.95 when aqueous foam moderation is assumed.

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 675 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMITS

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	500 system heatup and cooldown cycles at rates ≤ 100°F/hr.	Heatup cycle - T_{avg} from ≤ 200°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.

6.1 RESPONSIBILITY

6.1.1 The Plant 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 Vice President Nuclear Energy shall be reissued to all station personnel on an annual basis.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for unit management and technical support shall be as shown in Figure 6.2-1.

UNIT STAFF

6.2.2 The unit organization shall be as shown in Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Reactor Operator shall be in the control room when fuel is in the reactor. In addition, while the 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. All CORE ALTERATIONS shall be observed by a licensed operator and supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation. The SRO in charge of fuel handling normally supervises from the control room and has the flexibility to directly supervise at either the refueling deck or the spent fuel pool.

e. A site Fire Brigade of at least five members shall be maintained onsite at all times.# The Fire Brigade shall not include the Shift Supervisor, the STA, nor the two other members of the minimum shift crew necessary for safe shutdown of the unit and any personnel required for other essential functions during a fire emergency.

f. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g., senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

#The health physics technician and Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed 2 hours, in order to accommodate unexpected absence, provided immediate action is taken to fill the required positions.

UNIT STAFF (Continued)

Adequate shift coverage shall be maintained without routine heavy use of overtime. The objective shall be to have operating personnel work a normal 8-hour day, 40-hour week while the plant is operating. However, in the event that unforeseen problems require substantial amounts of overtime to be used, or during extended periods of shutdown for refueling, major maintenance or major plant modification, on a temporary basis the following guidelines shall be followed:

a. An individual should not be permitted to work more than 16 hours straight, excluding shift turnover time.

b. An individual should not be permitted to work more than 16 hours in any 24-hour period, nor more than 24 hours in any 48-hour period, nor more than 72 hours in any 7-day period, all excluding shift turnover time.

c. A break of at least 8 hours should be allowed between work periods, including shift turnover time.

d. Except during extended shutdown periods, the use of overtime should be considered on an individual basis and not for the entire staff on a shift.

Any deviation from the above guidelines shall be authorized by the Plant Manager or his deputy, or higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation. Controls shall be included in the procedures such that individual overtime shall be reviewed monthly by the Plant Manager or his designee to assure that excessive hours have not been assigned. Routine deviation from the above guidelines is not authorized.

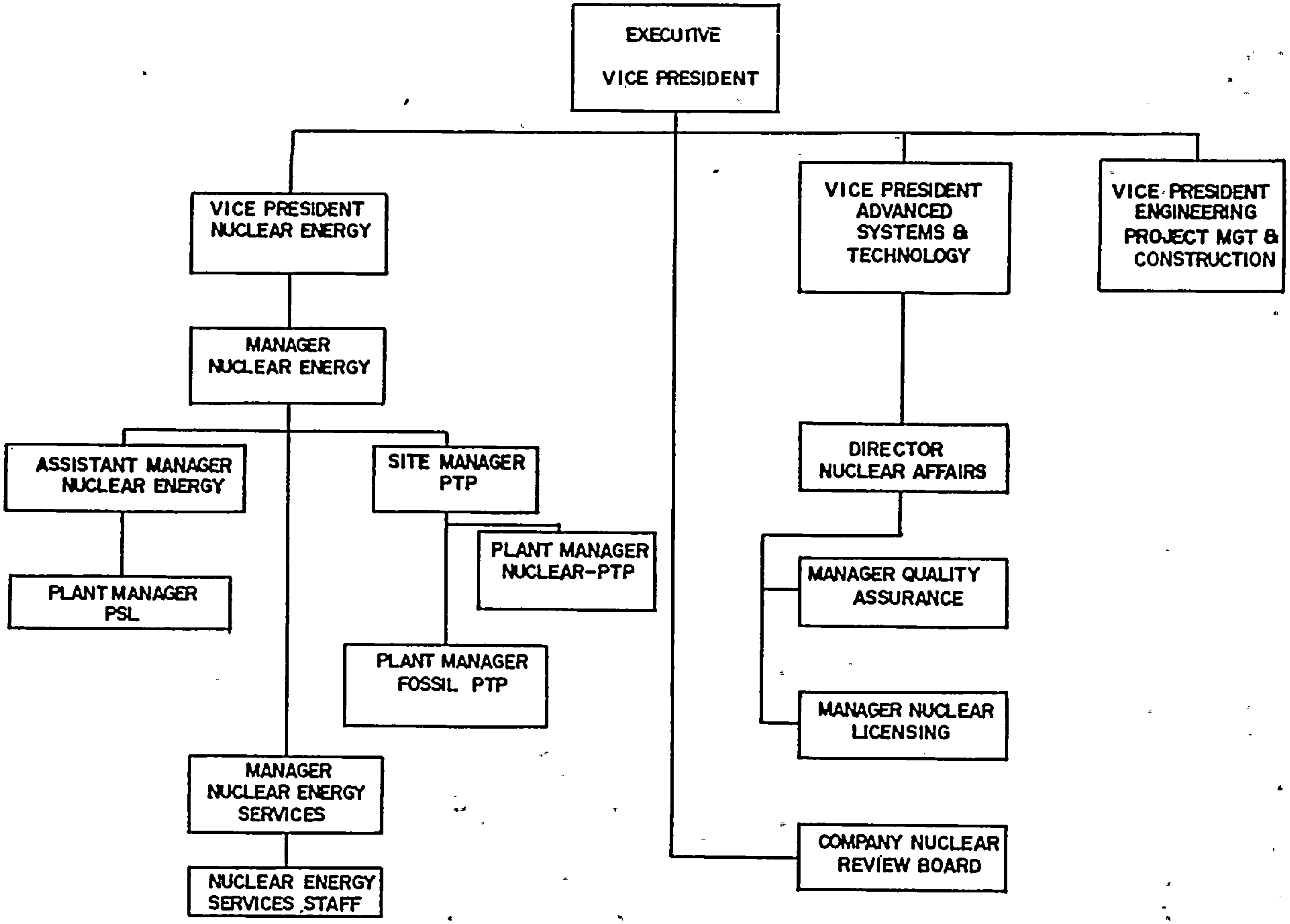


Figure 6.2-1
Offsite organization for facility management and technical support

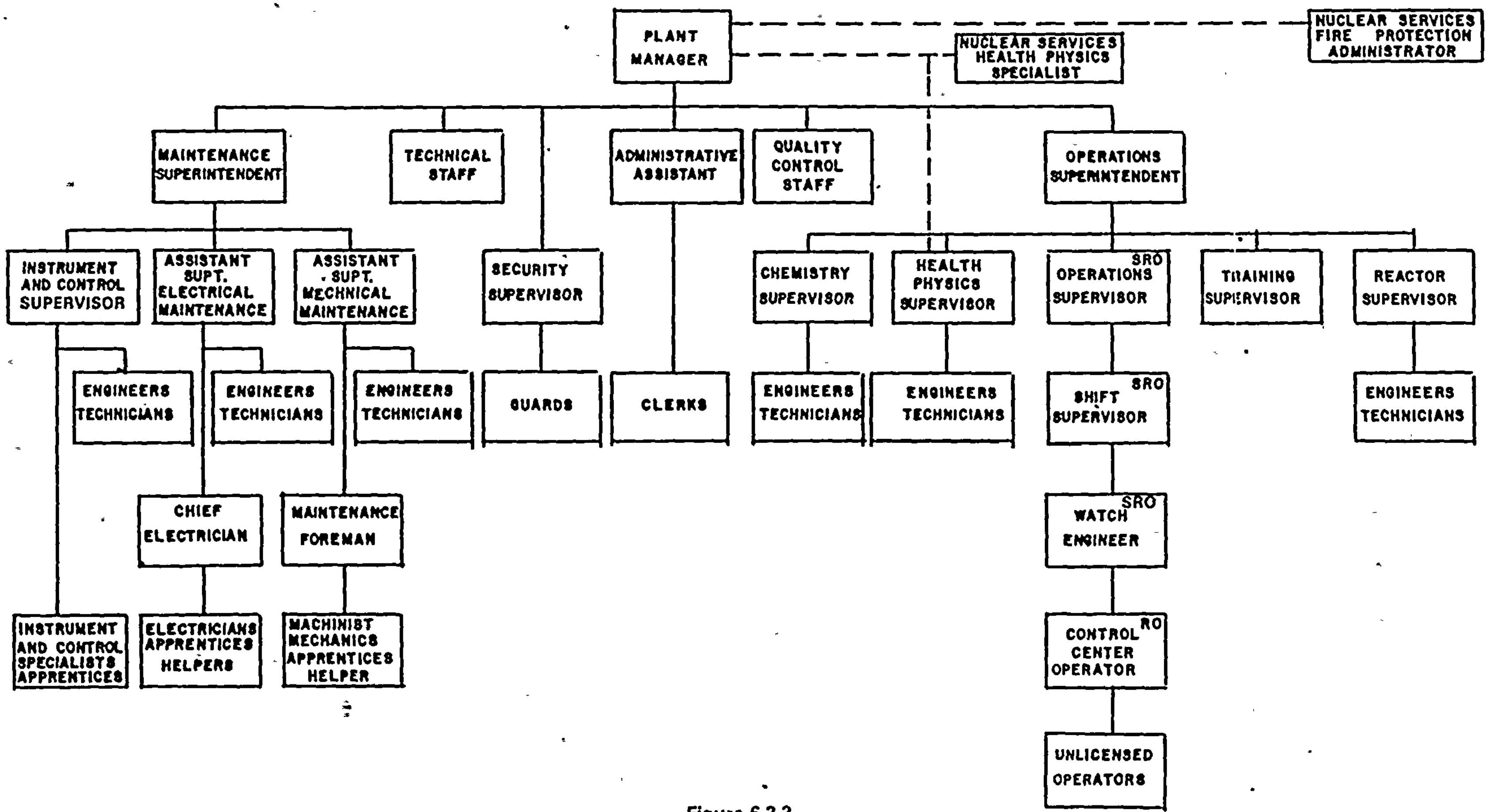


Figure 6.2-2
Unit organization St. Lucie Plant, Unit 2

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[a]	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.

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.

6.2.3 INDEPENDENT SAFETY ENGINEERING GROUP (ISEG)

FUNCTION

6.2.3.1 The ISEG shall function to examine plant operating characteristics, NRC issuances, industry advisories, Licensee Event Reports and other sources of plant design and operating experience information, including plants of similar design, which may indicate areas for improving plant safety.

COMPOSITION

6.2.3.2 The ISEG shall be composed of five dedicated, full-time members with varied backgrounds and disciplines related to nuclear power plants. No more than two members shall be assigned from any one department. Three or more of the members shall be engineers with a bachelor's degree in engineering or a related science, with at least 2 years of professional level experience in the nuclear field. Any nondegreed ISEG members will either be licensed as a Reactor Operator or Senior Reactor Operator, or will have been previously licensed as a Reactor Operator or Senior Reactor Operator within the last year at the St. Lucie Plant site; or they will meet the qualifications of a department head as specified in Specification 6.3.1 of the St. Lucie Unit 2 Technical Specifications. The qualifications of each nondegreed candidate for the ISEG shall be approved by the Assistant Chief Engineer - Power Plant Engineering, prior to joining the group.

RESPONSIBILITIES

6.2.3.3 The ISEG shall be responsible for maintaining surveillance of selected plant activities to provide independent verification* that these activities are performed correctly and that human errors are reduced as much as practical. The ISEG shall make detailed recommendations for revised procedures, equipment modifications, maintenance activities, operations activities, or other means of improving plant safety to the Assistant Chief Engineer-Power Plant Engineering.

AUTHORITY

6.2.3.4 The ISEG is an onsite independent technical review group that reports offsite to the Assistant Chief Engineer-Power Plant Engineering. The ISEG shall have the authority necessary to perform the functions and responsibilities as delineated above.

RECORDS

6.2.3.5 Records of activities performed by the ISEG shall be prepared, maintained and a report of the activities forwarded each calendar month to the Assistant Chief Engineer-Power Plant Engineering.

6.2.4 SHIFT TECHNICAL ADVISOR

The Shift Technical Advisor shall provide advisory technical support to the Shift Supervisor in the areas of thermal hydraulics, reactor engineering, and plant analysis with regard to the safe operation of the unit.

6.3 UNIT STAFF QUALIFICATIONS

6.3.1 Each member of the unit staff shall meet or exceed the minimum qualifications of ANSI/ANS-3.1-1978 as endorsed by Regulatory Guide 1.8, September 1975 (reissued May 1977), except for the (1) Health Physics Supervisor who shall meet

*Not responsible for sign-off function.

6.3 UNIT STAFF QUALIFICATIONS (Continued)

or exceed the qualifications of Regulatory Guide 1.8, September 1975 and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and plant operating characteristics, including transients and accidents.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the unit staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI/ANS 3.1-1978 and Appendix "A" of 10 CFR Part 55 and the supplemental requirements specified in Sections A and C of Enclosure 1 of the March 28, 1980 NRC letter to all licensees, and shall include familiarization with relevant industry operational experience.

6.4.2 Fire drills will be conducted at least quarterly for all Fire Brigade personnel.

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 Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Facility Review Group shall be composed of the:

Member:	Plant Manager
Member:	Operations Superintendent
Member:	Operations Supervisor
Member:	Maintenance Superintendent or Assistant Supt. Mechanical Maintenance
Member:	Reactor Supervisor
Member:	Health Physics Supervisor or Chemistry Supervisor
Member:	Technical Supervisor
Member:	Quality Control Supervisor
Member:	Assistant Supt. Electrical Maintenance or Instrument and Control Supervisor

The Chairman shall be a member of the FRG and shall be designated in writing.

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 procedures required by Specification 6.8 and changes thereto, (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 Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix A Technical Specifications.

d. Review of all proposed changes or modifications to unit systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications, including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager, Nuclear Energy, Vice President Nuclear Energy, and to the Chairman of the Company Nuclear Review Board.

f. Review of events requiring 24-hour written notification to the Commission.

g. Review of unit operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Plant Manager or the Company Nuclear Review Board.

i. Review of the Security Plan and implementing procedures and submittal of recommended changes to the Company Nuclear Review Board.

j. Review of the Emergency Plan and implementing procedures and submittal of recommended changes to the Company Nuclear Review Board.

RESPONSIBILITIES (Continued)

k. Review of every unplanned onsite 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 Vice President Nuclear Energy 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.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend in writing to the Plant Manager, prior to implementation except as provided in Specification 6.8.3, approval or disapproval of items considered under Specifications 6.5.1.6a. through d. and m. above.

b. Render determinations in writing, prior to implementation except as provided in Specification 6.8.3, with regard to whether or not each item considered under Specifications 6.5.1.6a. through e. above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President Nuclear Energy and the Company Nuclear Review Board of disagreement between the FRG and the Plant Manager; however, the Plant 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 Vice President Nuclear Energy 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. instrumentation and control

f. radiological safety

g. mechanical and electrical engineering

h. quality assurance practices

COMPOSITION

6.5.2.2 The CNRB shall be composed of the following members:

Member:	Vice President, Advanced Systems and Technology
Member:	Chief Engineer, Power Plant Engineering
Member:	Vice President, Nuclear Energy
Member:	Manager, Nuclear Energy
Member:	Director of Nuclear Affairs
Member:	Manager, Nuclear Fuel
Member:	Power Plant Engineering Principal Engineer
Member:	Power Plant Engineering Senior Project Manager

The Chairman shall be a member of the CNRB and shall be designated in writing.

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 CNBR 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 four 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 safety evaluations for (1) changes to procedures, equipment, or systems and (2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment, or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications or this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of unit equipment that affect nuclear safety.

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of structures, systems, or components that could affect nuclear safety.

i. Reports and meetings minutes of the 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 at least once per 12 months.

b. The performance, training and qualifications of the entire unit staff at least once per 12 months.

c. The results of actions taken to correct deficiencies occurring in unit equipment, structures, systems, or method of operation that affect nuclear safety at least once per 6 months.

AUDITS (Continued)

d. The performance of activities required by the Quality Assurance Program to meet the criteria of Appendix B, 10 CFR Part 50, at least once per 24 months.

e. Any other area of unit operation considered appropriate by the CNRB or the Executive Vice President.

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 at least once per 12 months.

i. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

j. The PROCESS CONTROL PROGRAM and implementing procedures for dewatering of radioactive bead resin at least once per 24 months.

AUTHORITY

6.5.2.9 The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specified in Specifications 6.5.2.7 and 6.5.2.8.

RECORDS

6.5.2.10 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 Executive Vice President 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 Executive Vice President within 14 days following completion of the review.

c. Audit reports encompassed by Specification 6.5.2.8 above shall be forwarded to the Executive Vice President and to the management positions responsible for the areas audited within 30 days after completion of the audit by the auditing organization.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24-hour notification to the Commission shall be reviewed by the FRG and the results of this review submitted to the CNRB and the Vice President Nuclear Energy.

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 Vice President Nuclear Energy 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 Vice President Nuclear Energy 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. Security Plan implementation.

e. Emergency Plan implementation.

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 Each procedure of Specification 6.8.1a. through i. above, and changes thereto, shall be reviewed by the FRG and shall be approved by the Plant Manager prior to implementation and shall be reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of Specification 6.8.1a. through i. above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the FRG and approved by the Plant Manager within 14 days of implementation.

6.8.4 The following programs shall be established, implemented, maintained, and shall be audited under the cognizance of the CNRB at least once per 24 months:

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 for 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. Post-accident Sampling*

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

(i) Training of personnel,

(ii) Procedures for sampling and analysis, and

(iii) Provisions for maintenance of sampling and analysis equipment.

*The post-accident sampling system (PASS) shall be completely installed and OPERABLE prior to initial criticality.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Regional Administrator of the Regional Office of the NRC unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation) supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS[1/]

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include:

a. A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors) receiving exposures

[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 whole body dose received from external sources should be assigned to specific major work functions.

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 Director, Office of Resource Management, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, with a copy to the Regional Administrator of the Regional Office of the NRC, no later than the 15th of each month following the calendar month covered by the report.

REPORTABLE OCCURRENCES

6.9.1.7 The REPORTABLE OCCURRENCES of Specifications 6.9.1.8 and 6.9.1.9 below, including corrective actions and measures to prevent recurrence, shall be reported to the NRC. Supplemental reports may be required to fully describe final resolution of occurrence. In case of corrected or supplemental reports, a licensee event report shall be completed and reference shall be made to the original report date.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP

6.9.1.8 The types of events listed below shall be reported within 24 hours by telephone and confirmed by telegraph, mailgram, or facsimile transmission to the Regional Administrator of the Regional Office of the NRC or his designate no later than the first working day following the event, with a written followup report within 14 days. The written followup report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

[2/] This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

PROMPT NOTIFICATION WITH WRITTEN FOLLOWUP (Continued)

a. Failure of the reactor protection system or other systems subject to Limiting Safety System Settings to initiate the required protective function by the time a monitored parameter reaches the setpoint specified as the limiting safety system setting in the technical specifications or failure to complete the required protective function.

b. Operation of the unit or affected systems when any parameter or operation subject to a limiting condition for operation is less conservative than the least conservative aspect of the Limiting Condition for Operation established in the Technical Specifications.

c. Abnormal degradation discovered in fuel cladding, reactor coolant pressure boundary, or primary containment.

d. Reactivity anomalies involving disagreement with the predicted value of reactivity balance under steady-state conditions during power operation greater than or equal to 1% $\Delta k/k$; a calculated reactivity balance indicating a SHUTDOWN MARGIN less conservative than specified in the Technical Specifications; short-term reactivity increases that correspond to a reactor period of less than 5 seconds or, if subcritical, an unplanned reactivity insertion of more than 0.5% $\Delta k/k$; or occurrence of any unplanned criticality.

e. Failure or malfunction of one or more components which prevents or could prevent, by itself, the fulfillment of the functional requirements of system(s) used to cope with accidents analyzed in the FSAR.

f. Personnel error or procedural inadequacy which prevents or could prevent, by itself, the fulfillment of the functional requirements of systems required to cope with accidents analyzed in the FSAR.

g. Conditions arising from natural or man-made events that, as a direct result of the event require unit shutdown, operation of safety systems, or other protective measures required by Technical Specifications.

h. Errors discovered in the transient or accident analyses or in the methods used for such analyses as described in the safety analysis report or in the bases for the Technical Specifications that have or could have permitted reactor operation in a manner less conservative than assumed in the analyses.

i. Performance of structures, systems, or components that requires remedial action or corrective measures to prevent operation in a manner less conservative than assumed in the accident analyses in the safety analysis report or Technical Specifications bases; or discovery during unit life of conditions not specifically considered in the safety analysis report or Technical Specifications that require remedial action or corrective measures to prevent the existence or development of an unsafe condition.

j. Offsite releases of radioactive materials in liquid and gaseous effluents that exceed the limits of Specification 3.11.1.1 or 3.11.2.1.

k. Exceeding the limits in Specification 3.11.2.6 for the storage of radioactive materials in the listed tanks. The written followup report shall include a schedule and a description of activities planned and/or taken to reduce the contents to within the specified limits.

THIRTY DAY WRITTEN REPORTS

6.9.1.9 The types of events listed below shall be the subject of written reports to the Regional Administrator of The Regional Office of the NRC within 30 days of occurrence of the event. The written report shall include, as a minimum, a completed copy of a licensee event report form. Information provided on the licensee event report form shall be supplemented, as needed, by additional narrative material to provide complete explanation of the circumstances surrounding the event.

a. Reactor protection system or engineered safety features instrument settings which are found to be less conservative than those established by the Technical Specifications but which do not prevent the fulfillment of the functional requirements of affected systems.

b. Conditions leading to operation in a degraded mode permitted by a Limiting Condition for Operation or plant shutdown required by a Limiting Condition for Operation.

c. Observed inadequacies in the implementation of administrative or procedural controls which threaten to cause reduction of degree of redundancy provided in reactor protection systems or engineered safety features systems.

d. Abnormal degradation of systems other than those specified in Specification 6.9.1.8c above designed to contain radioactive material resulting from the fission process.

e. An unplanned offsite release of (1) more than 1 curie of radioactive material in liquid effluents, (2) more than 150 curies of noble gas in gaseous effluents, or (3) more than 0.05 curie of radioiodine in gaseous effluents. The report of an unplanned offsite release of radioactive material shall include the following information:

1. A description of the event and equipment involved.

2. Cause(s) for the unplanned release.

3. Actions taken to prevent recurrence.

4. Consequences of the unplanned release.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.10 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents form Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

Every 2 years using the previous 6 months release history for isotopes and historical meteorolgical data determine the controlling age group for both liquid and gaseous pathways. If changed from current submit change to ODCM to reflect new tables for these groups and use the new groups in subsequent dose calculations.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases for the previous

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*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.

**In lieu of submission with the first half year Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (Continued)

calendar year. Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, March 1976.

The Radioactive Effluent Release Reports shall include the following information for each class of solid waste (as defined by 10 CFR Part 61) shipped offsite during the report period:

a. Volume,

b. Total curie quantity (specify whether determined by measurement or estimate),

c. Principal radionuclides (specify whether determined by measurement or estimate),

d. Type of waste (e.g., dewatered spent resin, compacted dry waste, evaporator bottoms)

e. Type of container (e.g., LSA, Type A, Type B, Large Quantity), and

f. Solidification agent or absorbent (e.g., cement, urea formaldehyde).

The Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to UNRESTRICTED AREAS of radioactive materials in gaseous and liquid effluents made during the reporting period.

The Radioactive Effluent Release Reports shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), as well as a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to Specification 3.12.2.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT*

6.9.1.11 Routine Radiological Environmental Operating Reports covering the operation of the unit during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality.

The Annual Radiological Environmental Operating Reports shall include summaries, interpretations, and information based on trend analysis of the results of the radiological environmental surveillance activities for the report period, including a comparison, as appropriate, with preoperational studies, with operational controls and with previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by Specification 3.12.2.

The Annual Radiological Environmental Operating Reports shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the

*A single submittal may be made for a multiple unit station.

ANNUAL RADIOLOGICAL ENVIRONMENTAL OPERATING REPORT (Continued)

locations specified in the Table and Figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as possible in a supplementary report.

The reports shall also include the following: a summary description of the radiological environmental monitoring program; at least two legible maps* covering all sampling locations keyed to a table giving distances and directions from the centerline of one reactor; the results the Interlaboratory Comparison Program, required by Specification 3.12.3: discussion of all deviations from the sampling schedule of Table 3.12-1; and discussion of all analyses in which the LLD required by Table 4.12-1 was not achievable.

6.9.1.12 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 Regional Administrator of the Regional Office of the NRC.

6.9.1.13 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 Regional Administrator of the Regional Office of the NRC.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of 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 OCCURRENCES submitted to the Commission.

d. Records of surveillance activities; inspections and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

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*One map shall cover stations near the SITE BOUNDARY; a second shall include the more distant stations.

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 inservice 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 listed in Tables 3.7-4a and 3.7-4b including the date at which the service life commences and associated installation and maintenance records.

m. Records of secondary water sampling and water quality.

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.

6.11 RADIATION PROTECTION PROGRAM

Procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20 and shall be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

6.12 HIGH RADIATION AREA

6.12.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR Part 20, each high radiation area in which the intensity of radiation is greater than 100 mrem/hr but less than 1000 mrem/hr shall be barricaded and conspicuously posted as a high radiation area and entrance thereto shall be controlled by requiring issuance of a Radiation Work Permit (RWP)*. 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 such that a major portion of the body could receive in one hour a dose greater than 1000 mrem 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 such that a major portion of the body could receive in one hour a dose in excess of 1000 mrem** 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.

**Measurement made at 18 inches from source of radioactivity.

6.13 PROCESS CONTROL PROGRAM (PCP)

6.13.1 The PCP shall be approved by the Commission prior to implementation.

6.13.2 Licensee initiated changes to the PCP:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

   a. Sufficiently detailed information to totally suport the rationale for the change without benefit of additional or supplemental information;

   b. A determination that the change did not reduce the overall conformance of the dewatered bead resin to existing criteria for radioactive wastes; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.14 OFFSITE DOSE CALCULATION MANUAL (ODCM)

6.14.1 The ODCM shall be approved by the Commission prior to implementation.

6.14.2 Licensee initiated changes to the ODCM:

1. Shall be submitted to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made effective. This submittal shall contain:

   a. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with an approval and date box, together with appropriate analyses or evaluations justifying the change(s);

   b. A determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. Documentation of the fact that the change has been reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

6.15 MAJOR CHANGES TO RADIOACTIVE LIQUID, GASEOUS AND SOLID WASTE TREATMENT SYSTEMS*

6.15.1 Licensee initiated major changes to the radioactive waste systems (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the Facility Review Group. The discussion of each shall contain:

   a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.

   b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

   c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

   d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

   e. An evaluation of the change which shows the expected maximum exposures to individual in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto;

   f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

   g. An estimate of the exposure to plant operating personnel as a result of the change; and

   h. Documentation of the fact that the change was reviewed and found acceptable by the FRG.

2. Shall become effective upon review and acceptance by the FRG.

*Licensees may chose to submit the information called for in this Specification as part of the annual FSAR update.

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16. ABSTRACT *(200 words or less)*

The St. Lucie Plant, Unit 2, Technical Specifications were prepared by the U. S. Nuclear Regulatory Commission to set forth the limits, operating conditions, and other requirements applicable to a nuclear reactor facility as set forth in Section 50.36 of 10 CFR Part 50 for the protection of the health and safety of the public.

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