ST. LUCIE PLANT

UNIT 1

TECHNICAL SPECIFICATIONS

APPENDIX "A"

TO

LICENSE NO. DPR-67

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SECTION 1.0

DEFINITIONS

DEFINED TERMS

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

THERMAL POWER

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

RATED THERMAL POWER

1.3 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2560 MWt.

OPERATIONAL MODE

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

ACTION

1.5 ACTION shall be those additional requirements specified as corollary statements to each principle specification and shall be part of the specifications.

OPERABLE - OPERABILITY

1.6 A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s). Implicit in this definition shall be the assumption that all necessary attendant instrumentation, controls, electric 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).

REPORTABLE OCCURRENCE

1.7 A REPORTABLE OCCURRENCE shall be any of those conditions specified in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

CONTAINMENT VESSEL INTEGRITY

1.8 CONTAINMENT VESSEL INTEGRITY shall exist when:

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

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

b. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed position except as provided in Table 3.6-2 of Specification 3.6.3.1,

1.8.2 All containment vessel equipment hatches are closed and sealed,

1.8.3 Each containment vessel airlock is OPERABLE pursuant to Specification 3.6.1.3, and

1.8.4 The containment leakage rates are within the limits of Specification 3.6.1.2.

CHANNEL CALIBRATION

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

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be 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.

CORE ALTERATION

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

SHUTDOWN MARGIN

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

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

b. No change in part length control element assembly position.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

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

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

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

UNIDENTIFIED LEAKAGE

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

PRESSURE BOUNDARY LEAKAGE

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

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be the water flow from the reactor coolant pump seals.

AZIMUTHAL POWER TILT - T_q

1.18 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.

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

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same 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.

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$ 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.

STAGGERED TEST BASIS

1.21 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.

FREQUENCY NOTATION

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

AXIAL SHAPE INDEX

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

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

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_r^P

1.24 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.

SHIELD BUILDING INTEGRITY

1.25 SHIELD BUILDING INTEGRITY shall exist when:

1.25.1 Each door is closed except when the access opening is being used for normal transit entry and exit, and

1.25.2 The shield building ventilation system is OPERABLE.

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.

ENGINEERED SAFETY FEATURE RESPONSE TIME

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

PHYSICS TESTS

1.28 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.

TABLE 1.1

OPERATIONAL MODES

MODE	REACTIVITY CONDITION, K_{eff}	%RATED THERMAL POWER*	AVERAGE COOLANT TEMPERATURE
1. POWER OPERATION	≥ 0.99	$> 5\%$	$\geq 300°F$
2. STARTUP	≥ 0.99	$\leq 5\%$	$\geq 300°F$
3. HOT STANDBY	< 0.99	0	$\geq 300°F$
4. HOT SHUTDOWN	< 0.99	0	$300°F > T_{avg} > 200°F$
5. COLD SHUTDOWN	< 0.99	0	$\leq 200°F$
6. REFUELING**	≤ 0.95	0	$\leq 140°F$

* Excluding decay heat.

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

TABLE 1.2

FREQUENCY NOTATION

NOTATION	FREQUENCY
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 6 months.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.

SECTION 2.0

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

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

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of maximum cold leg temperature and THERMAL POWER has exceeded the appropriate pressurizer pressure line, be in HOT STANDBY within 1 hour.

REACTOR COOLANT SYSTEM PRESSURE

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

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

ACTION:

MODES 1 and 2

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

MODES 3, 4 and 5

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

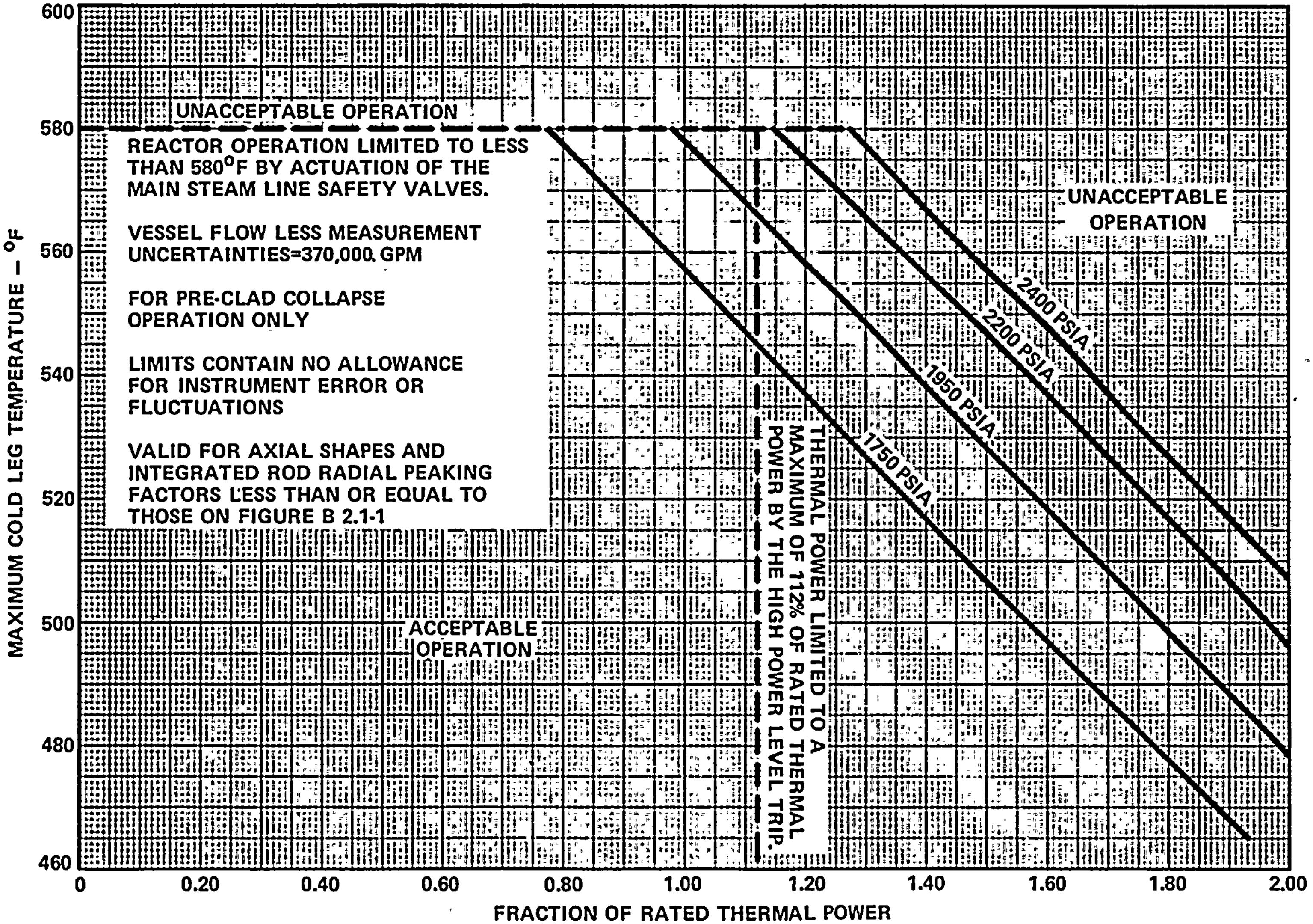

Figure 2.1-1 REACTOR CORE THERMAL MARGIN SAFETY LIMIT – FOUR REACTOR COOLANT PUMPS OPERATING

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 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High (1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 106.5% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 106.5% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	≥ 95% of design reactor coolant flow with 4 pumps operating*	≥ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 3.9 psig	≤ 3.9 psig
6. Steam Generator Pressure - Low (2)	≥ 485 psig	≥ 485 psig
7. Steam Generator Water Level -Low	≥ 36.3% Water Level - each steam generator	≥ 36.3% Water Level - each steam generator
8. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

*Design reactor coolant flow with 4 pumps operating is 390,000 gpm.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	$\geq$ 800 psig	$\geq$ 800 psig
11. Rate of Change of Power - High (4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 5% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 585 psig; bypass shall be automatically removed at or above 585 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

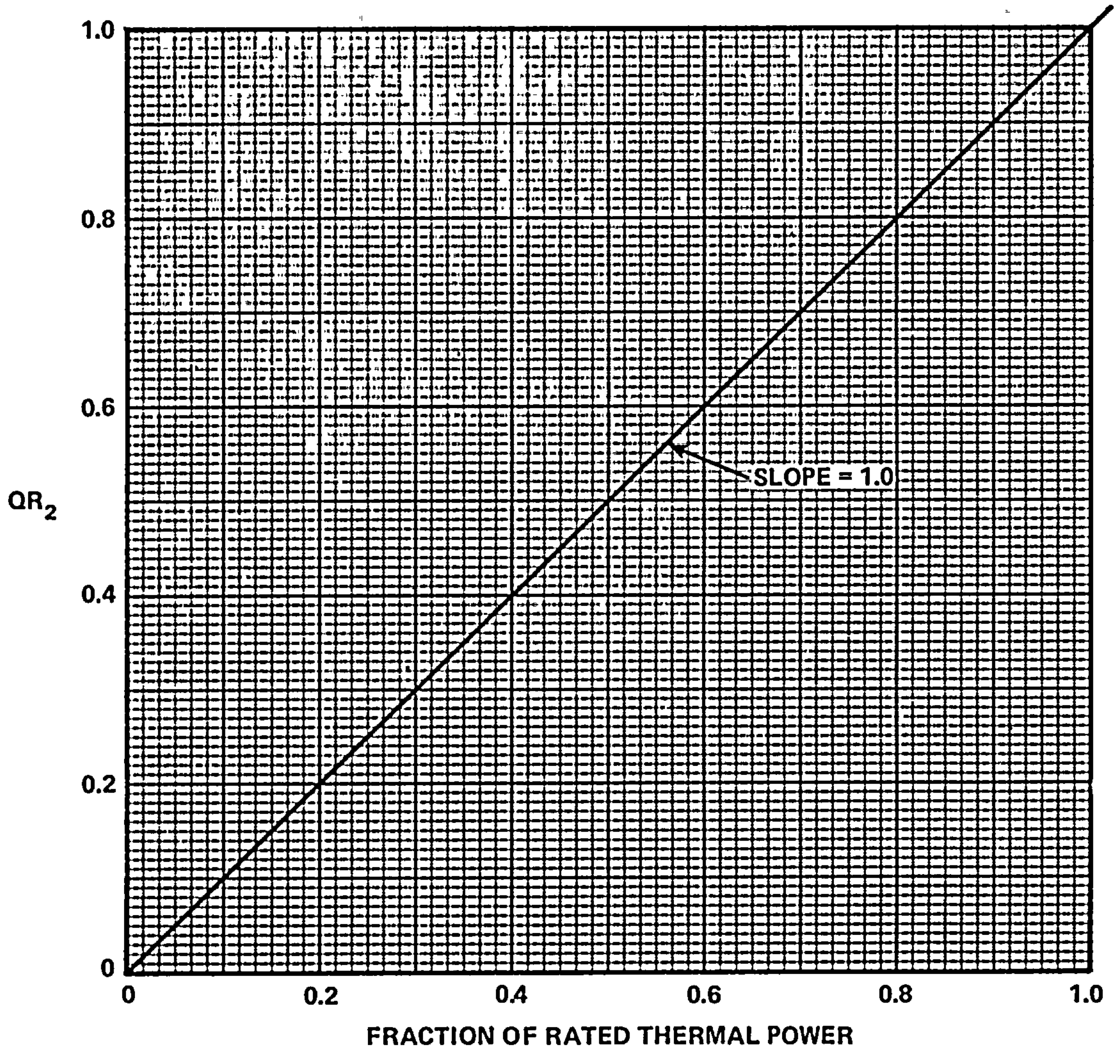

FIGURE 2.2-1

Local Power Density — High Trip Setpoint
Part 1 (Fraction of RATED THERMAL POWER Versus QR_2)

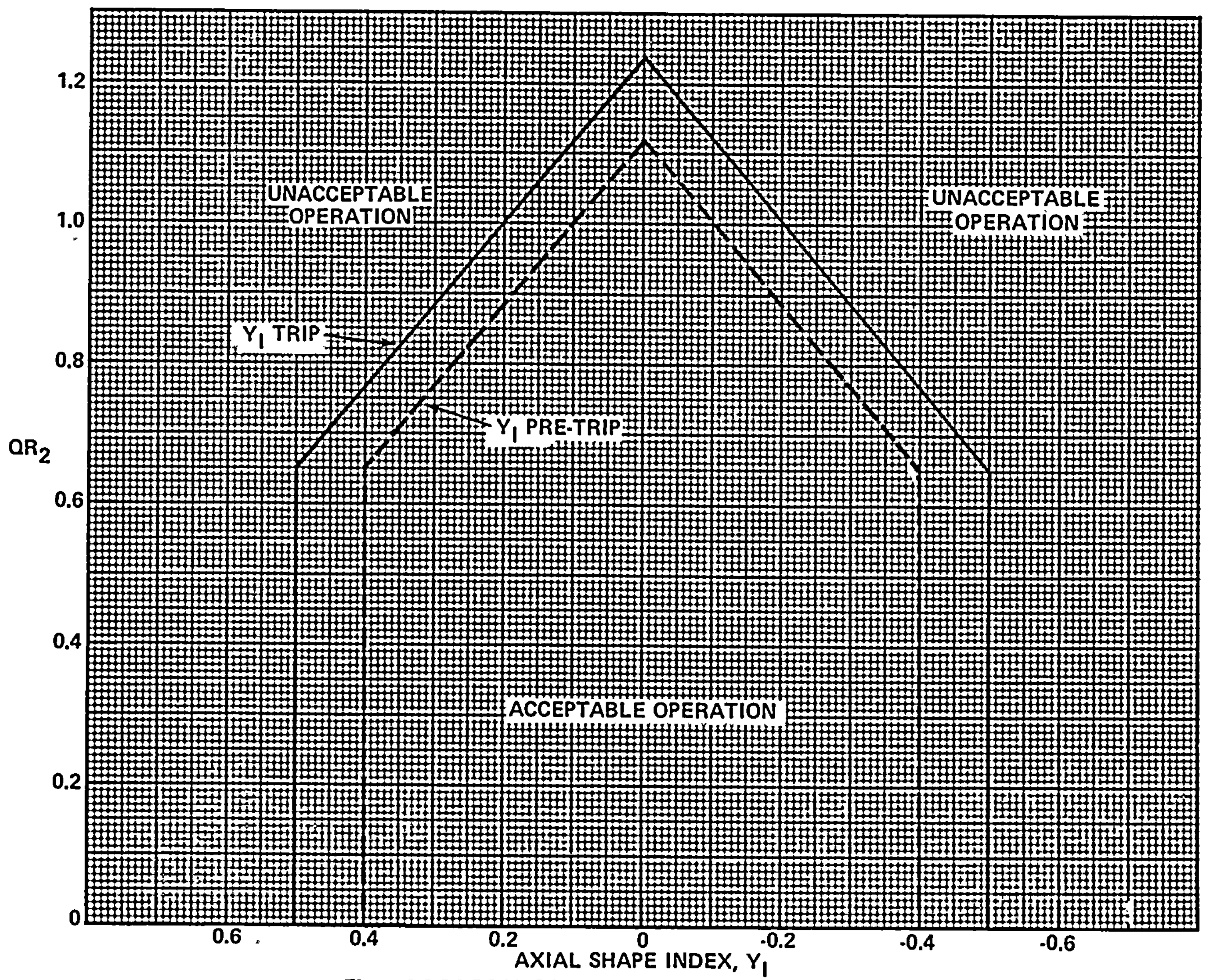

Figure 2.2-2 LOCAL POWER DENSITY – HIGH TRIP SETPOINT
PART 2 (QR_2 VERSUS Y_I)

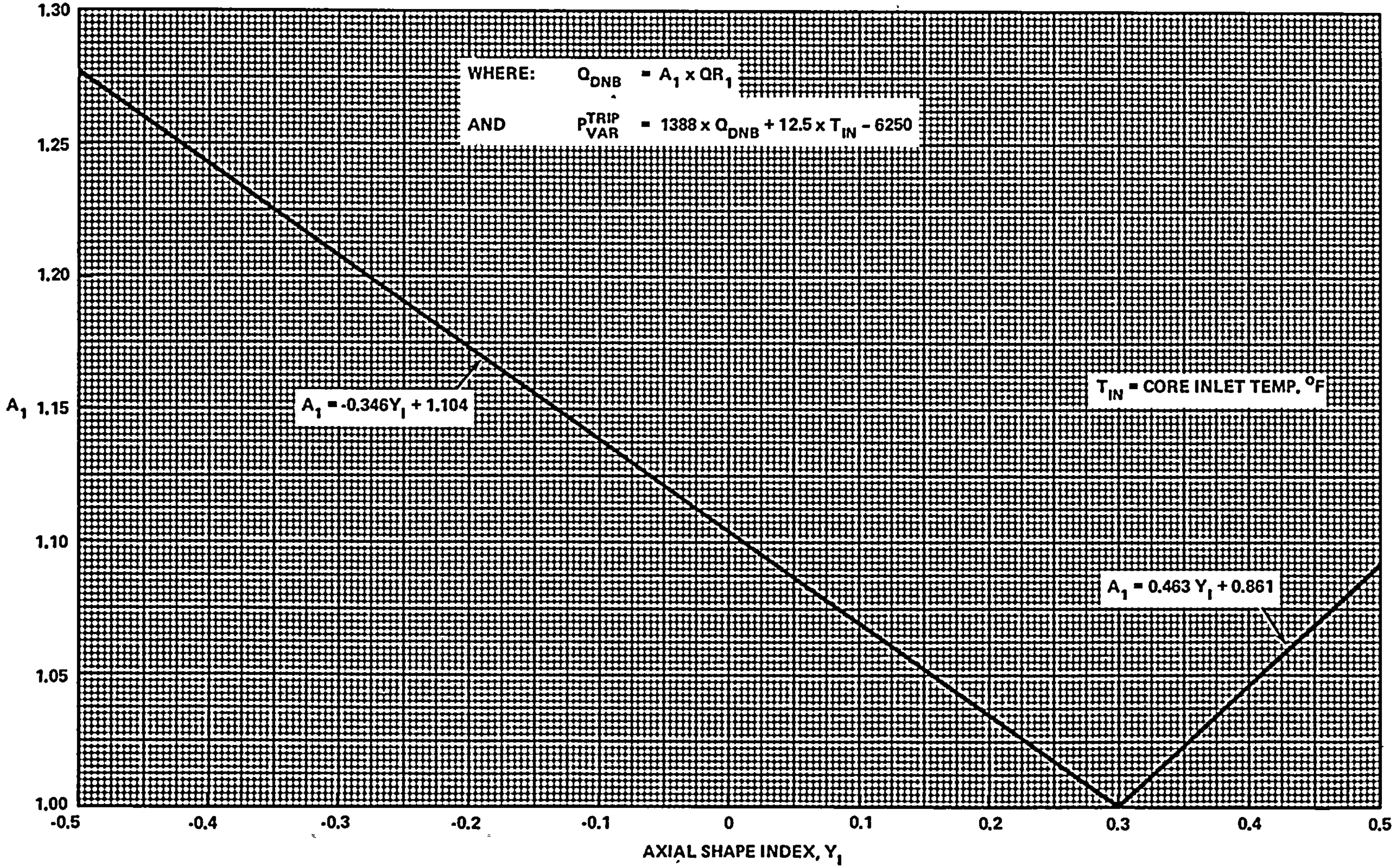

FIGURE 2.2-3
Thermal Margin/Low Pressure Trip Setpoint
Part 1 (Y_I Versus A_1)

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

AND $P_{VAR}^{TRIP} = 1388 \times Q_{DNB} + 12.5\ T_{IN} - 6250$

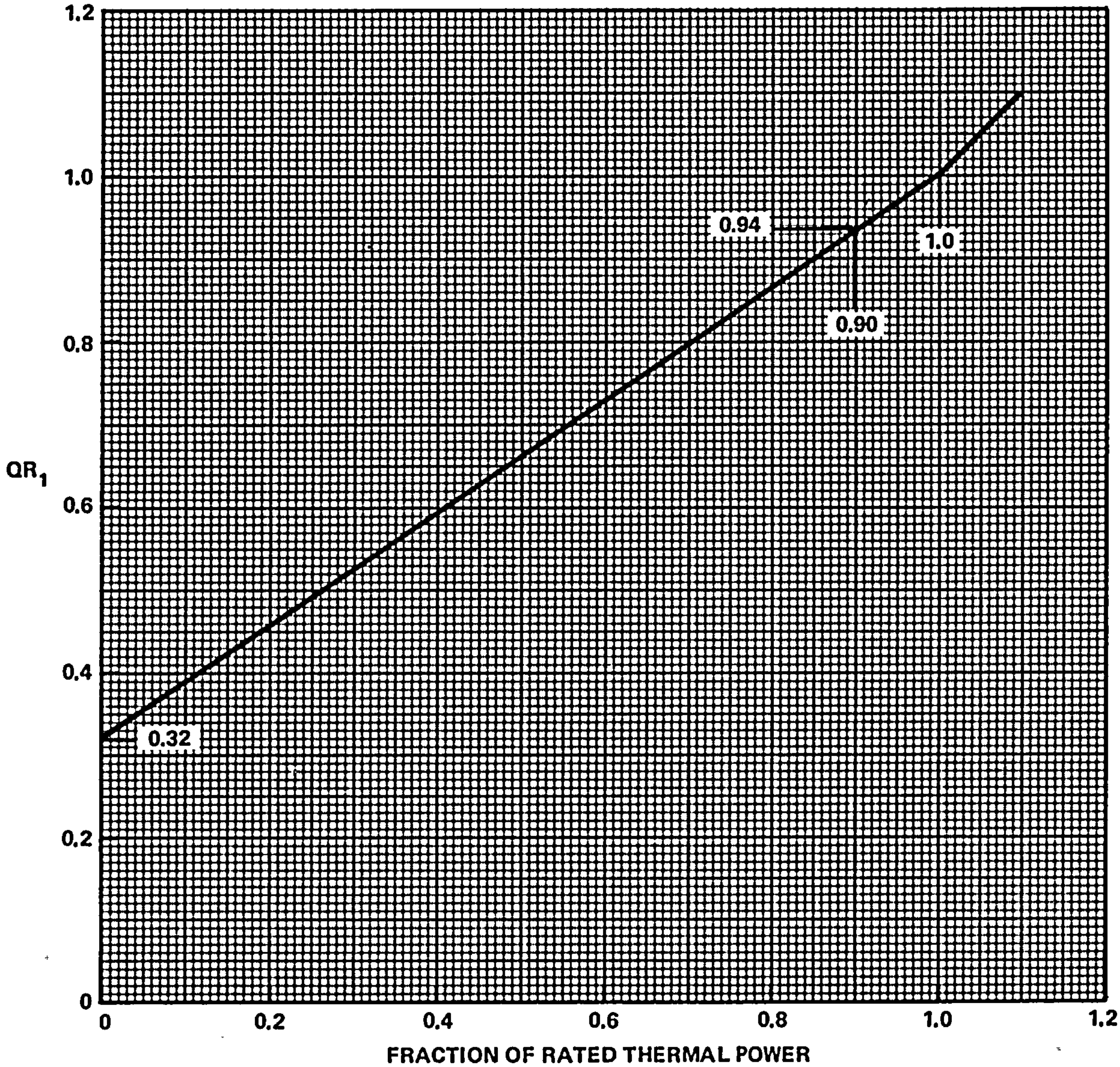

FIGURE 2.2-4

Thermal Margin/Low Pressure Trip Setpoint
Part 2 (Fraction of RATED THERMAL POWER Versus QR_1)

BASES

FOR

SECTION 2.0

SAFETY LIMITS

AND

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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 hear rate at or less than 21 kw/ft. Centerline fuel melting will not occur for this peak linear heat rate. 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 W-3 correlation. The W-3 DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent 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 1.30 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 2.1-1. The area of safe operation is below and to the left of these lines.

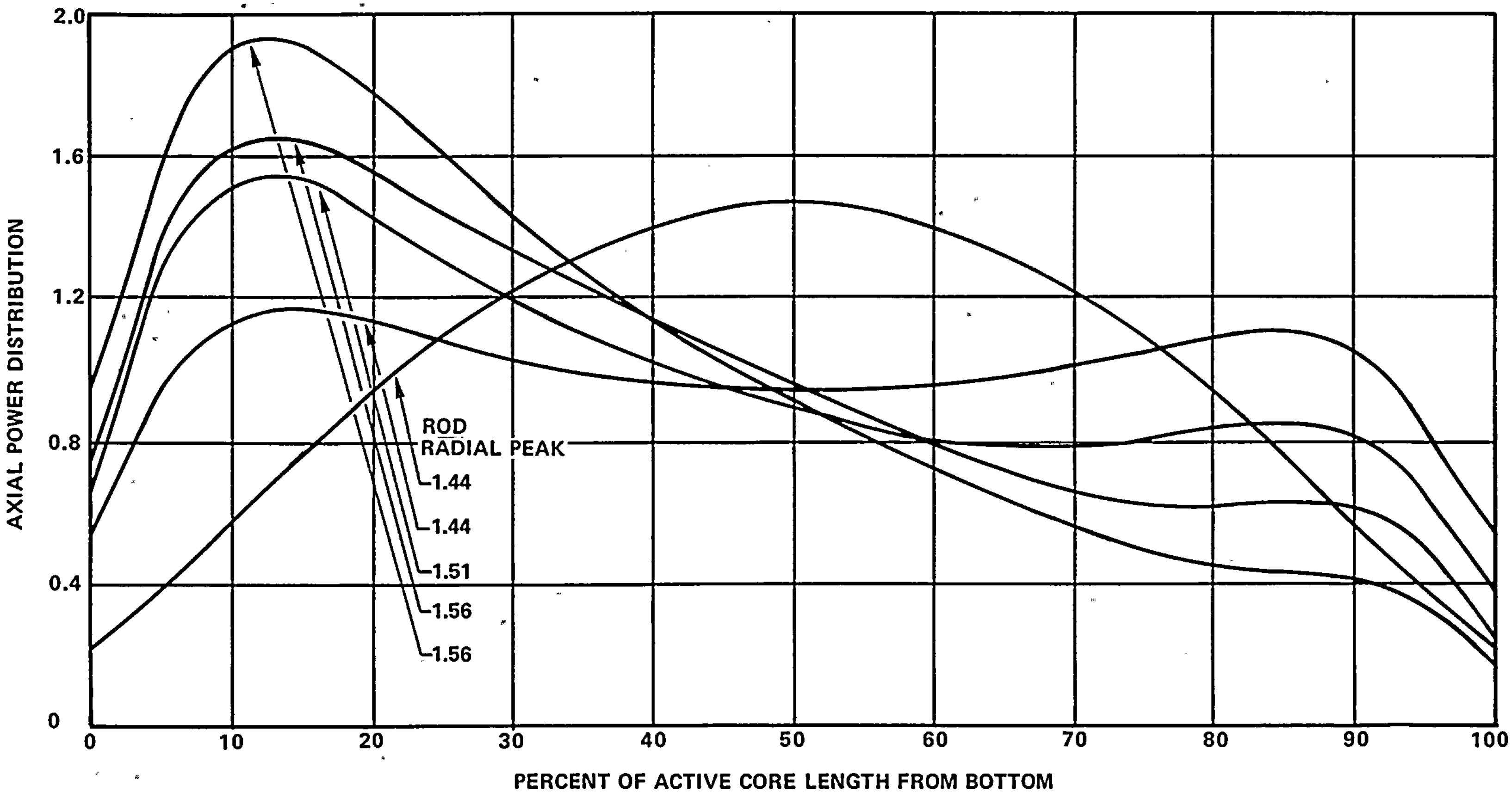

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

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

The reactor protective system in combination with the Limiting Conditions for Operation, is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than 1.30 and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

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

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

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

2.2.1 REACTOR TRIP SETPOINTS

The Reactor Trip Setpoints specified in Table 2.2-1 are the values at which the Reactor Trips are set for each parameter. The Trip Values have been selected to ensure that the reactor core and reactor coolant system are prevented from exceeding their safety limits. Operation with a trip set less conservative than its Trip Setpoint but within its specified Allowable Value is acceptable on the basis that each 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.

Power Level-High

The Power Level-High trip provides reactor core protection against reactivity excursions which are too rapid to be protected by a Pressurizer Pressure-High or Thermal Margin/Low Pressure trip.

The 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 106.5% of RATED THERMAL POWER and a minimum setpoint of 15% 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.

Reactor Coolant Flow-Low

The Reactor Coolant Flow-Low trip provides core protection to prevent DNB in the event of a sudden significant decrease in reactor coolant flow. Provisions have been made in the reactor protective system to permit operation of the reactor at reduced power if one or two

Reactor Coolant Flow-Low (Continued)

reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.30 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.30 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip in initiated concurrently with a safety injection.

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 setting of 485 psig is sufficiently below the full-load operating point of 800 psig so as not

Steam Generator Pressure-Low (Continued)

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 ± 22 psi in the accident analyses.

Steam Generator Water Level

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The local Power Density-High trip is provided to prevent the peak local power density in the fuel from exceeding 21 kw/ft during steady state operation thereby assuring that the melting point of the UO_2 fuel will not be reached. A value of 21 kw/ft is well below the value corresponding to fuel centerline melting.

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 ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent 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.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than 1.30, or when a void fraction limit is exceeded which could result in local flow instability.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1750 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 5% of RATED THERMAL POWER to compensate for potential power measurement error; an allowance of 2°F to compensate for potential temperature measurement uncertainty; and a further allowance of 47 psia to compensate for pressure measurement error, trip system processing 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 47 psia allowance is made up of a 22 psia pressure measurement allowance, a 5 psia trip system processing allowance and a 20 psia time delay allowance.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuring transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

LIMITING SAFETY SYSTEM SETTINGS

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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 accident 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.

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

3/4.0 APPLICABILITY

LIMITING CONDITION FOR OPERATION

3.0.1 Limiting Conditions for Operation and ACTION requirements shall be applicable during the OPERATIONAL MODES or other conditions specified for each specification.

3.0.2 Adherence to the requirements of the Limiting Condition for Operation and/or associated ACTION within the specified time interval shall constitute compliance with the specification. In the event the Limiting Condition for Operation is restored prior to expiration of the specified time interval, completion of the ACTION statement is not required.

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

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

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 test interval, and

SURVEILLANCE REQUIREMENTS (Continued)

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

4.0.3 Performance of a Surveillance Requirement within the specified time interval shall constitute compliance with OPERABILITY requirements for a Limiting Condition for Operation and associated ACTION statements unless otherwise required by the specification.

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 2.45% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 2.45% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 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 $\geq$ 2.45% $\Delta k/k$:

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

b. When in MODES 1 or 2#, 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##, at least once during CEA withdrawal and at least once per hour thereafter until the reactor is critical.

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.

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*See Special Test Exception 3.10.1.

#With $K_{eff} \geq 1.0$.

##With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, 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.

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

SHUTDOWN MARGIN - $T_{avg} \leq 200°F$

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 1.0% $\Delta k/k$.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN < 1.0% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 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 $\geq$ 1.0% $\Delta k/k$:

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

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

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

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.3 The flow rate of reactor coolant to the reactor pressure vessel shall be $\geq$ 3000 gpm whenever a reduction in Reactor Coolant System boron concentation 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 $\geq$ 3000 gpm within one 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 $\geq$ 3000 gpm to the reactor pressure vessel.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.5×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.2×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.5×10^{-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 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.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

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 refueling.

b. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 520 ppm.

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

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.5 The Reactor Coolant System lowest operating loop temperature (T_{avg}) shall be $\geq$ 515°F when the reactor is critical.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a Reactor Coolant System operating loop temperature (T_{avg}) < 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 $\geq$ 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 temperature (T_{avg}) is < 525°F.

* See Special Test Exception 3.10.3

With $K_{eff} \geq 1.0$.

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:

a. A flow path from the boric acid makeup tank via either a boric acid 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, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one injection path is restored to OPERABLE status.

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:

1. Cycling each testable power operated or automatic valve in the flow path required for boron injection through at least one complete cycle of full travel, and

2. 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.

SURVEILLANCE REQUIREMENTS

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 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 make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; 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:

1. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

2. 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:

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least once complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation signal.

CHARGING PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 5 and 6.

ACTION:

With no charging pump or high pressure safety injection pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one of the required pumps is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.3 At least the above required charging pump or high pressure safety injection pump shall be demonstrated OPERABLE at least once per 31 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying pump operation for at least 15 minutes, and

c. Verifying that the pump is aligned to receive electrical power from an OPERABLE emergency bus.

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 HOT STANDBY within the next 6 hours; restore at least two charging pumps to OPERABLE status within the next 48 hours or be in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.4 At least two charging pumps shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS by:

a. Starting (unless already operating) each pump from the control room, and

b. Verifying that each pump operates for at least 15 minutes.

BORIC ACID PUMPS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one boric acid pump shall be OPERABLE if only the flow path through the boric acid pump in Specification 3.1.2.1a above, is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no boric acid pump OPERABLE as required to complete the flow path of Specification 3.1.2.1a, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one boric acid pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.1.2.5 At least the above required boric acid pump shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 75 psig, and

c. Verifying pump operation for at least 15 minutes.

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.6 At least the boric acid pump(s) in the boron injection flow path(s) required OPERABLE pursuant to Specification 3.1.2.2a shall be OPERABLE if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one boric acid pump required for the boron injection flow path(s) pursuant to Specification 3.1.2.2a inoperable, restore the boric acid pump 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

4.1.2.6 At least the above required boric acid pump(s) shall be demonstrated OPERABLE at least once per 7 days by:

a. Starting (unless already operating) the pump from the control room,

b. Verifying, that on recirculation flow, the pump develops a discharge pressure of $\geq$ 75 psig, and

c. Verifying pump operation for at least 15 minutes.

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 one associated heat tracing circuit with the tank contents in accordance with Figure 3.1-1.

b. The refueling water tank with:

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

2. A minimum boron concentration of 1720 ppm, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water sources OPERABLE, suspend all operations involving positive reactivity changes until at least one borated water source is restored to OPERABLE status.

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 water level 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 site ambient air temperature is < 40°F.

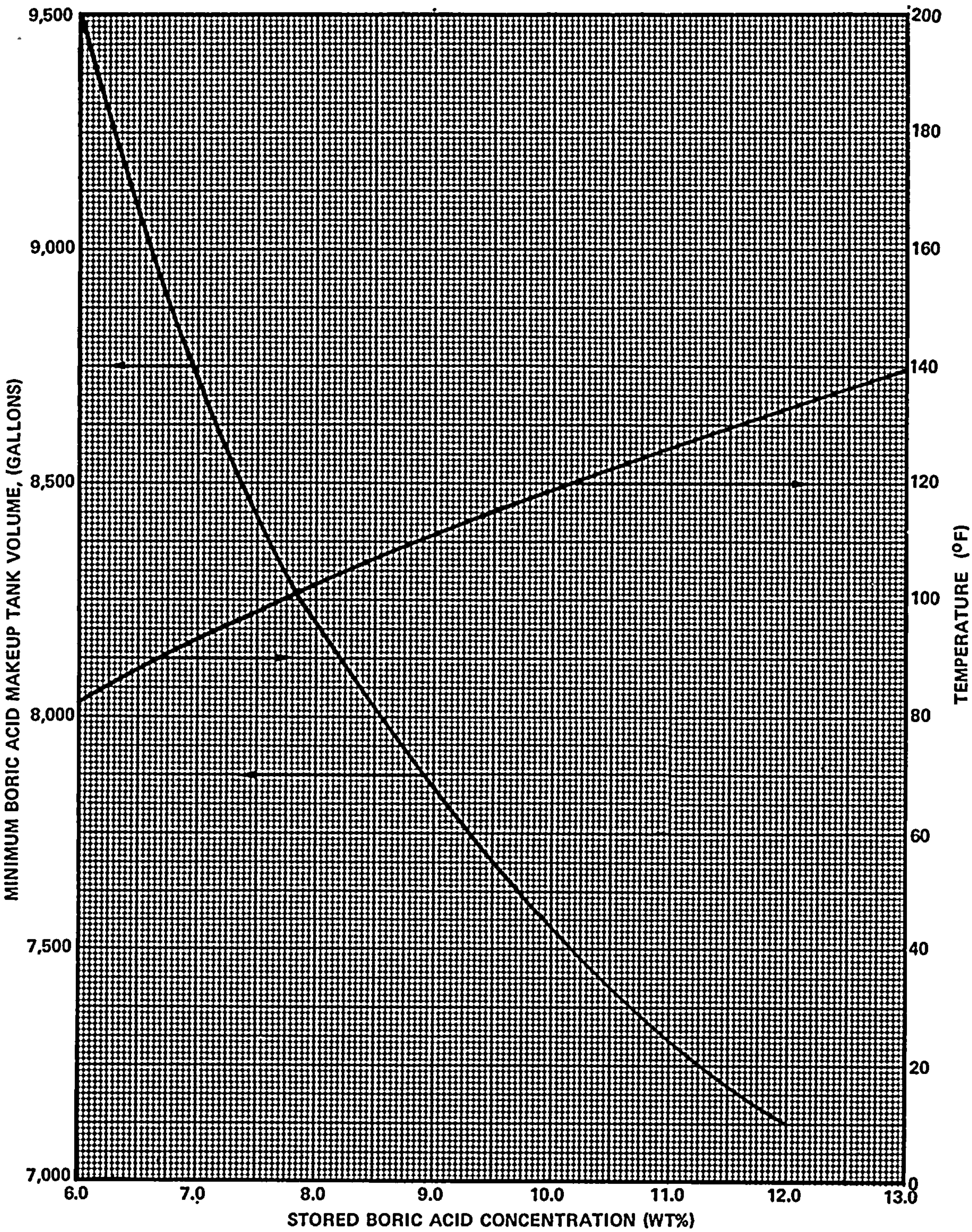


Figure 3.1-1 Minimum Boric Acid Makeup Tank Volume and Temperature as a Function of Stored Boric Acid Concentration

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. Two boric acid makeup tanks and one associated heat tracing circuit with the contents of the tanks in accordance with Figure 3.1-1, and

b. The refueling water tank with:

1. A minimum contained volume of 371,800 gallons of water,
2. A minimum boron concentration of 1720 ppm,
3. A maximum solution temperature of 100°F,
4. A minimum solution temperature of 55°F when in MODES 1 and 2, and
5. A minimum solution temperature of 40°F when in MODES 3 and 4.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one borated water source OPERABLE, restore at least two borated water sources to OPERABLE status within 72 hours or make the reactor subcritical within the next 2 hours and borate to a SHUTDOWN MARGIN equivalent to at least 1% $\Delta k/k$ at 200°F; restore at least two borated water sources to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

a. At least one per 7 days by:

1. Verifying the boron concentration in each water source,

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the water level in each water source, and

3. Verifying the boric acid makeup tank solution temperature.

b. At least once per 24 hours by verifying the RWT temperature.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

FULL LENGTH CEA POSITION

LIMITING CONDITION FOR OPERATION

3.1.3.1 The CEA Block Circuit and all full length (shutdown and regulating) CEAs shall be OPERABLE with each CEA of a given group positioned within 7.5 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, be in HOT STANDBY within 6 hours.

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

   1. Restore the CEA Block Circuit to OPERABLE status, or

   2. Place and maintain the CEA drive system mode switch in either the "Manual" or "Off" position and fully withdraw all CEAs in groups 3, 4, 5 and 6 and withdraw the CEAs in group 7 to less than 5% insertion, or

   3. Be in at least HOT STANDBY.

c. With one full length CEA inoperable (unless immovable as a result of excessive friction or mechanical interference or known to be untrippable) but within its above specified alignment requirements, operation in MODES 1 and 2 may continue for up to 7 days per occurrence with a total accumulated time of $\leq$ 14 days per calendar year.

d. With one or more full length CEAs misaligned from any other CEAs in its group by more than 7.5 inches but less than 15 inches, operation in MODES 1 and 2 may continue, provided that within one hour the misaligned CEA(s) is either:

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

*See Special Test Exceptions 3.10.2 and 3.10.5.

LIMITING CONDITION FOR OPERATION (Continued)

2. Declared inoperable. After declaring the CEA inoperable, operation in MODES 1 and 2 may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided all of the following conditions are met:

   a) The THERMAL POWER level shall be reduced to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination within one hour; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used.

   b) Within one hour after reducing the THERMAL POWER as required by a) above, the remainder of the CEAs in the group with the inoperable CEA shall be aligned to within 7.5 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.

e. With one full length CEA misaligned from any other CEA in its group by 15 inches or more, reduce THERMAL POWER to ≤ 70% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination within one hour; if negative reactivity insertion is required to reduce THERMAL POWER, boration shall be used. Within one hour after reducing THERMAL POWER as required above, either:

   1. Restore the CEA to within the above specified alignment requirements, or

   2. Declare the CEA inoperable. After declaring the CEA inoperable, POWER OPERATION may continue for up to 7 days per occurrence with a total accumulated time of ≤ 14 days per calendar year provided the remainder of the CEAs in the group with the inoperable CEA are aligned to within 7.5 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.

LIMITING CONDITION FOR OPERATION (Continued)

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

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length CEA shall be determined to be within 7.5 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 shall be determined to be OPERABLE by movement of at least 7.5 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 maintains the CEA 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.

PART LENGTH CEA INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All part length CEAs shall be withdrawn to at least 132.0 inches.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With a maximum of one PLCEA withdrawn to less than 132.0 inches, either:

a. Withdraw the PLCEA to at least 132.0 inches within one hour, or

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

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each part length CEA shall be determined withdrawn to at least 132.0 inches by:

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

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

*See Special Test Exception 3.10.2.

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All shutdown, regulating and part length 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 $\pm$ 2.25 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With one or more PLCEA reed switch or pulse counting position indicator channels inoperable and the applicable PLCEA fully withdrawn and electric power to its drive mechanism disconnected, operation may continue provided the applicable PLCEA is verified immediately and at least once per 12 hours thereafter to be fully withdrawn by its "Full Out" limit.

b. With a maximum of one reed switch position indicator channel per group or one (except as permitted by ACTION item d. 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 $\leq$ 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

LIMITING CONDITION FOR OPERATION

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.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the 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 this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable),

2. The fully inserted 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.

d. With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.3.3 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 4.5 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.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.3 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

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

b. For specifically affected individual 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 at least 132.0 inches.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

With a maximum of one shutdown CEA withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, to less than 132.0 inches, within one hour either:

a. Withdraw the CEA to at least 132.0 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 at least 132.0 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 $K_{eff} \geq 1.0$.

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 with CEA insertion between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits restricted to:

a. $\leq$ 4 hours per 24 hour interval,

b. $\leq$ 5 Effective Full Power Days per 30 Effective Full Power Day interval, and

c. $\leq$ 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 two hours either:

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

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

b. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits for intervals > 4 hours per 24 hour interval, except during operation pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, 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 $\leq$ 5% of RATED THERMAL POWER per hour.

*See Special Test Exceptions 3.10.2 and 3.10.5.

#With $K_{eff} \geq 1.0$.

REGULATING CEA INSERTION LIMITS (Continued)

LIMITING CONDITION FOR OPERATION

c. With the regulating CEA groups inserted between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits for intervals > 5 EFPD per 30 EFPD interval or > 14 EFPD per calendar year, except during operations pursuant to the provisions of ACTION items c. and d. of Specification 3.1.3.1, either:

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

2. Be in 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 between the Long Term Steady State Insertion Limits and the Power Dependent Insertion Limits shall be determined at least once per 24 hours.

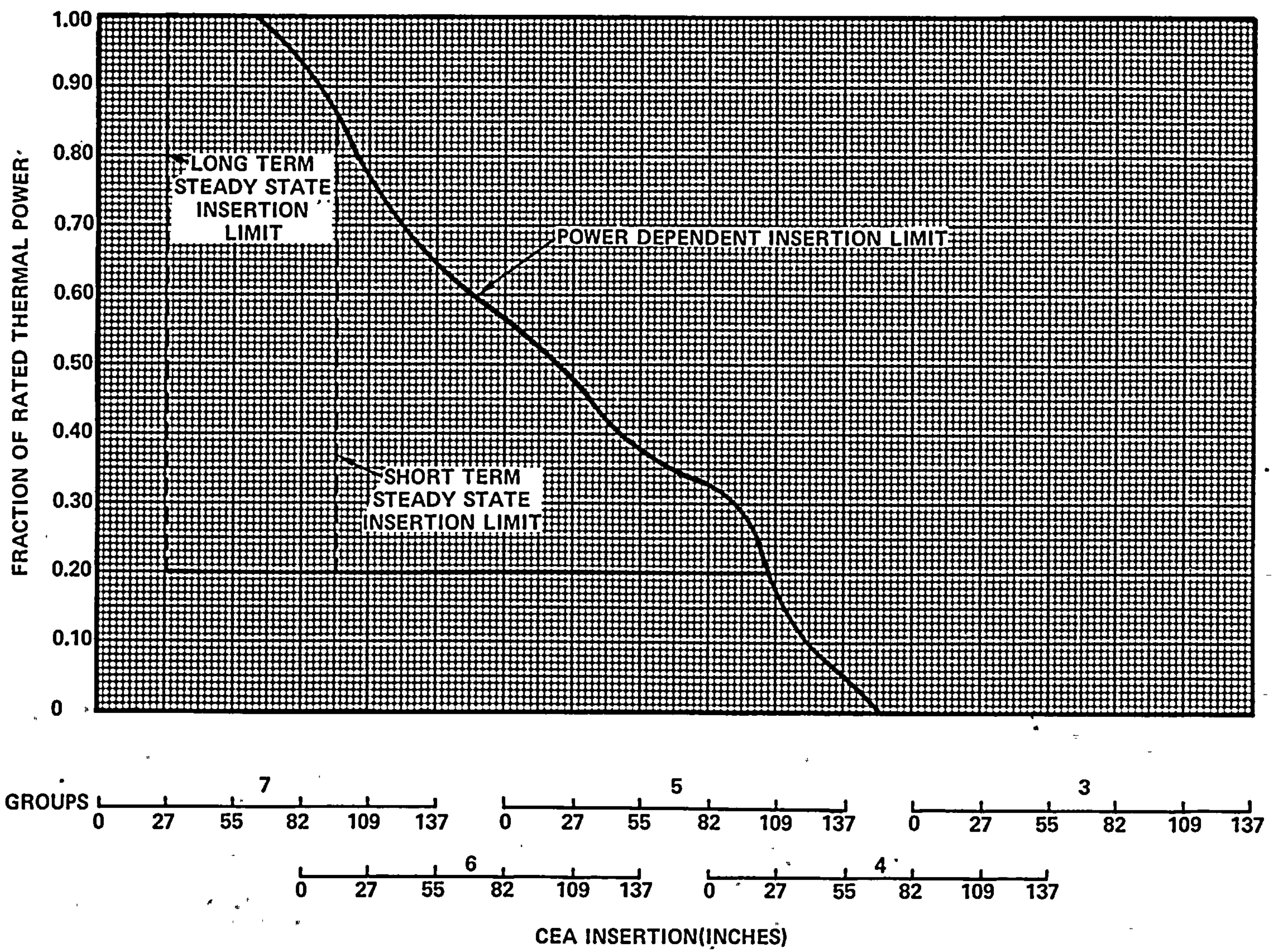


Figure 3.1-2 CEA Insertion Limits vs THERMAL POWER with 4 Reactor Coolant Pumps Operating

3/4.2 POWER DISTRIBUTION LIMITS

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 limits on the Power Ratio Recorder, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

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

b. Be in 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 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.

b. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the determination made in Specification 4.2.1.3.c, and

SURVEILLANCE REQUIREMENTS (Continued)

c. Verifying at least once per 31 days that the THERMAL POWER does not exceed the value determined by the following relationship:

$$\frac{L}{17.85} \times M$$

where:

1. L is the maximum allowable linear heat rate as determined from Figure 3.2-1 and is based on the core average burnup at the time of the latest incore flux map.

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

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.

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 1.10,

   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,

   5. A THERMAL POWER measurement uncertainty factor of 1.02, and

   6. A rod bow penalty factor of 1.05.

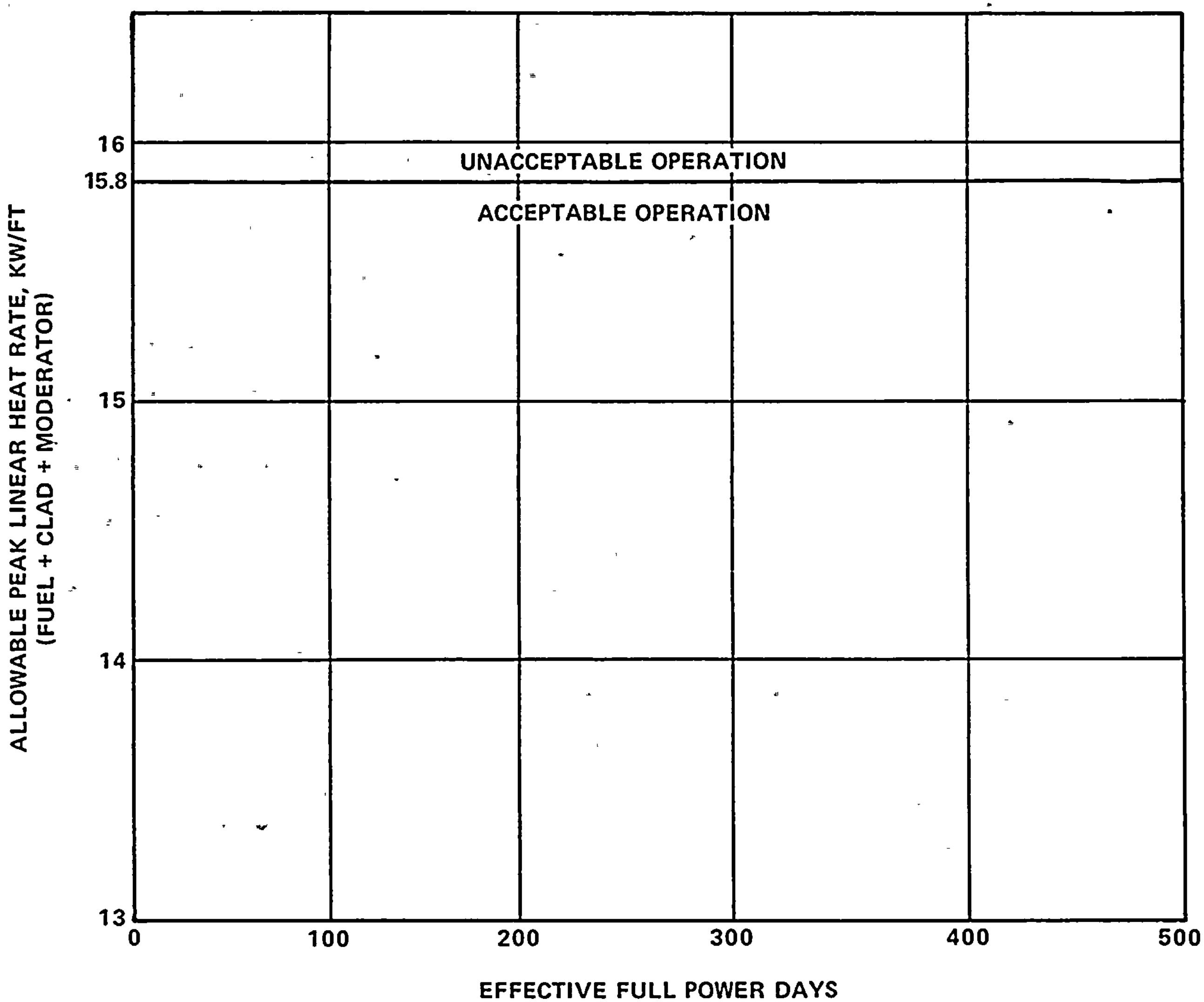


Figure 3.2-1 Allowable Peak Linear Heat Rate vs Burnup

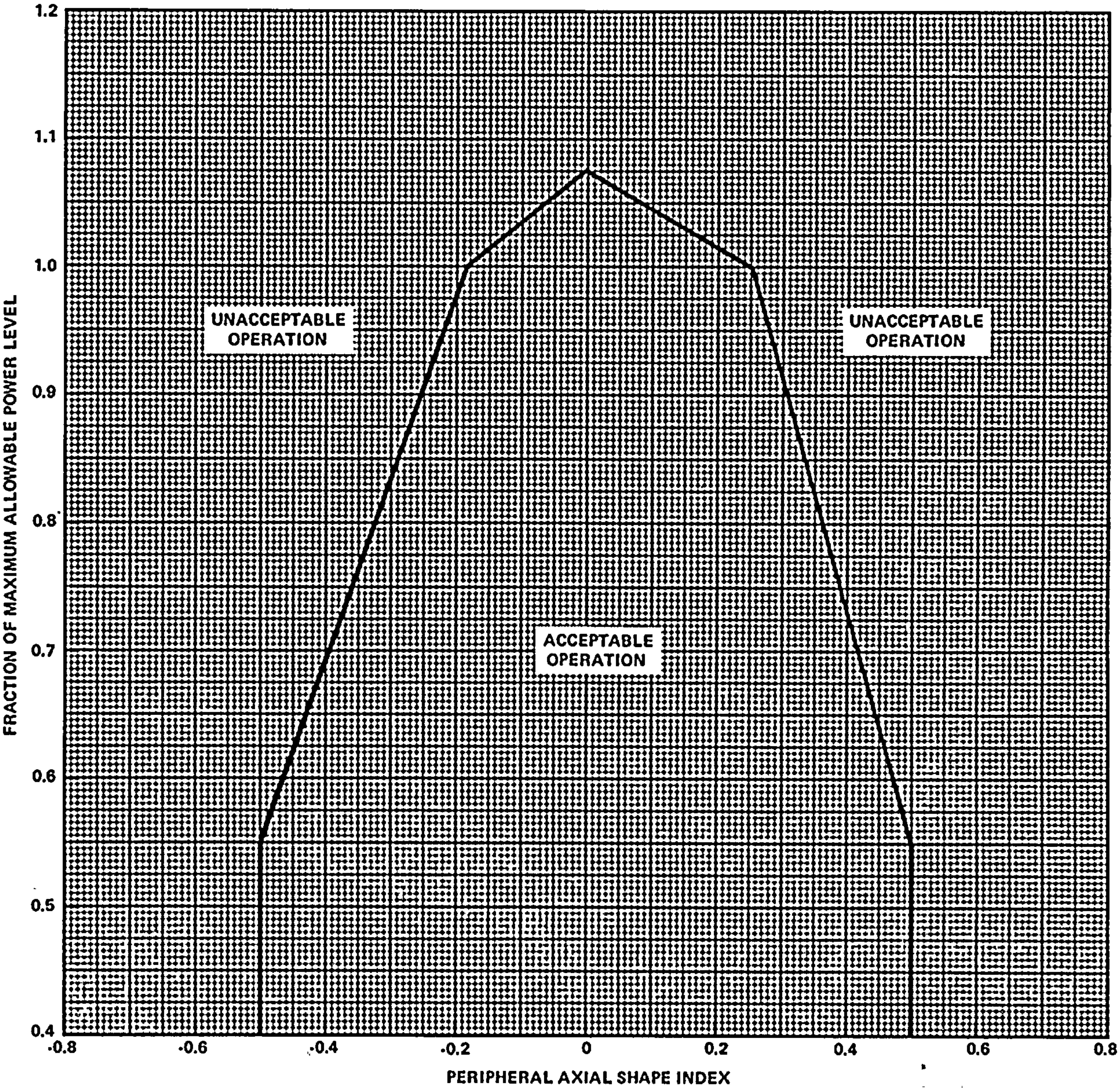


Figure 3.2-2

AXIAL SHAPE INDEX vs Fraction of Maximum Allowable Power Level per Specification 4.2.1.3

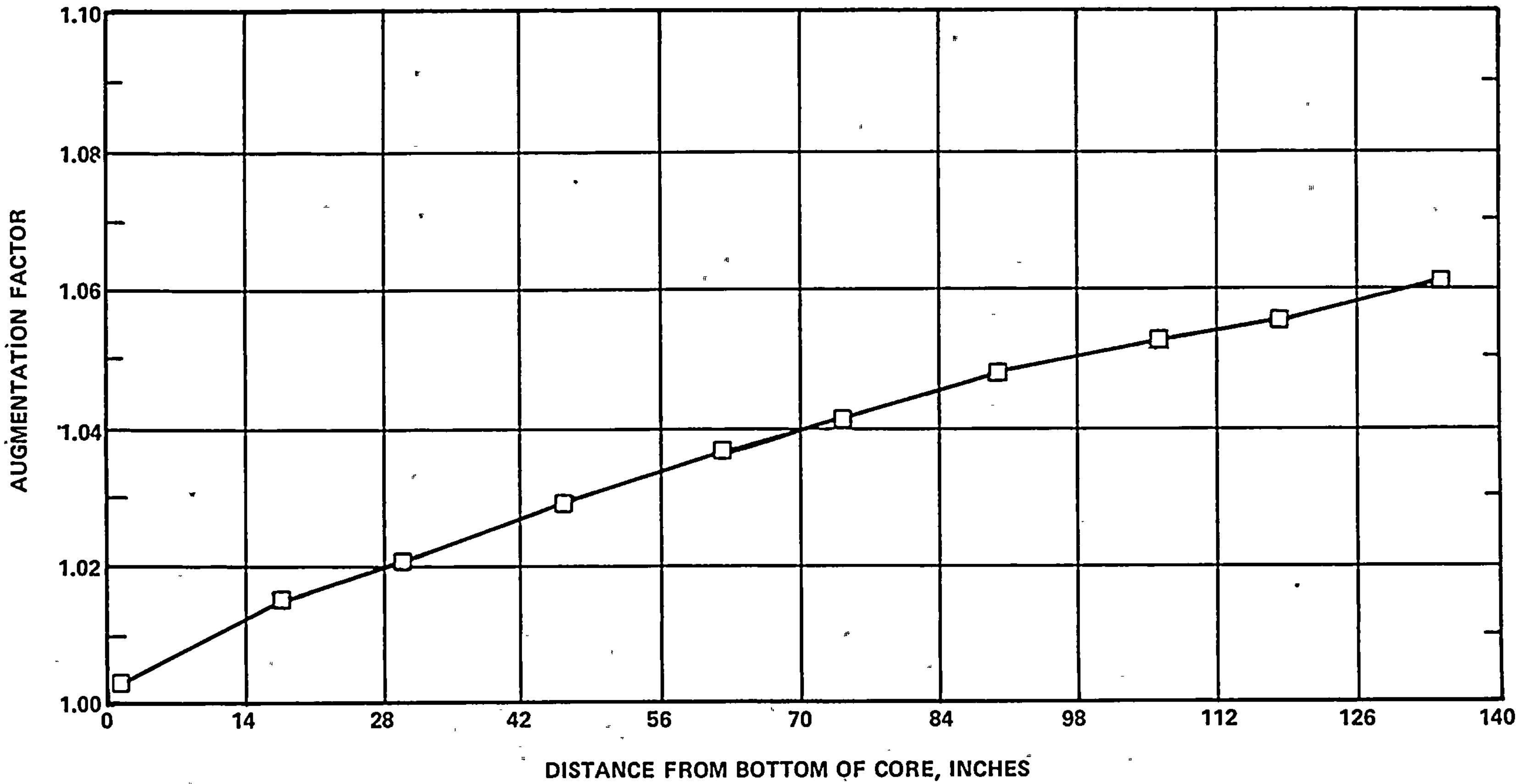


Figure 4.2-1 Augmentation Factors vs Distance from Bottom of Core

TOTAL RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F_r^T, defined as $F_r^T = F_r^P(1+T_q)$, shall be limited to $\leq$ 1.36.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.36, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to within the limits of Figure 3.2-3, fully withdraw the PLCEAS and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Reduce the Local Power Density - High and Thermal Margin/Low Pressure trip setpoints and the setpoints on the Power Ratio Calculator by a factor equivalent to $\geq \frac{F_r^T \text{ (meas)}}{1.36}$, or

c. 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_r^T shall be calculated by the expression $F_r^T = F_r^P(1+Tq)$ and F_r^T shall be determined to be within its limit at the following intervals:

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

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

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.02.

4.2.2.3 F_r^P 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 no part length CEAs inserted and 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_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

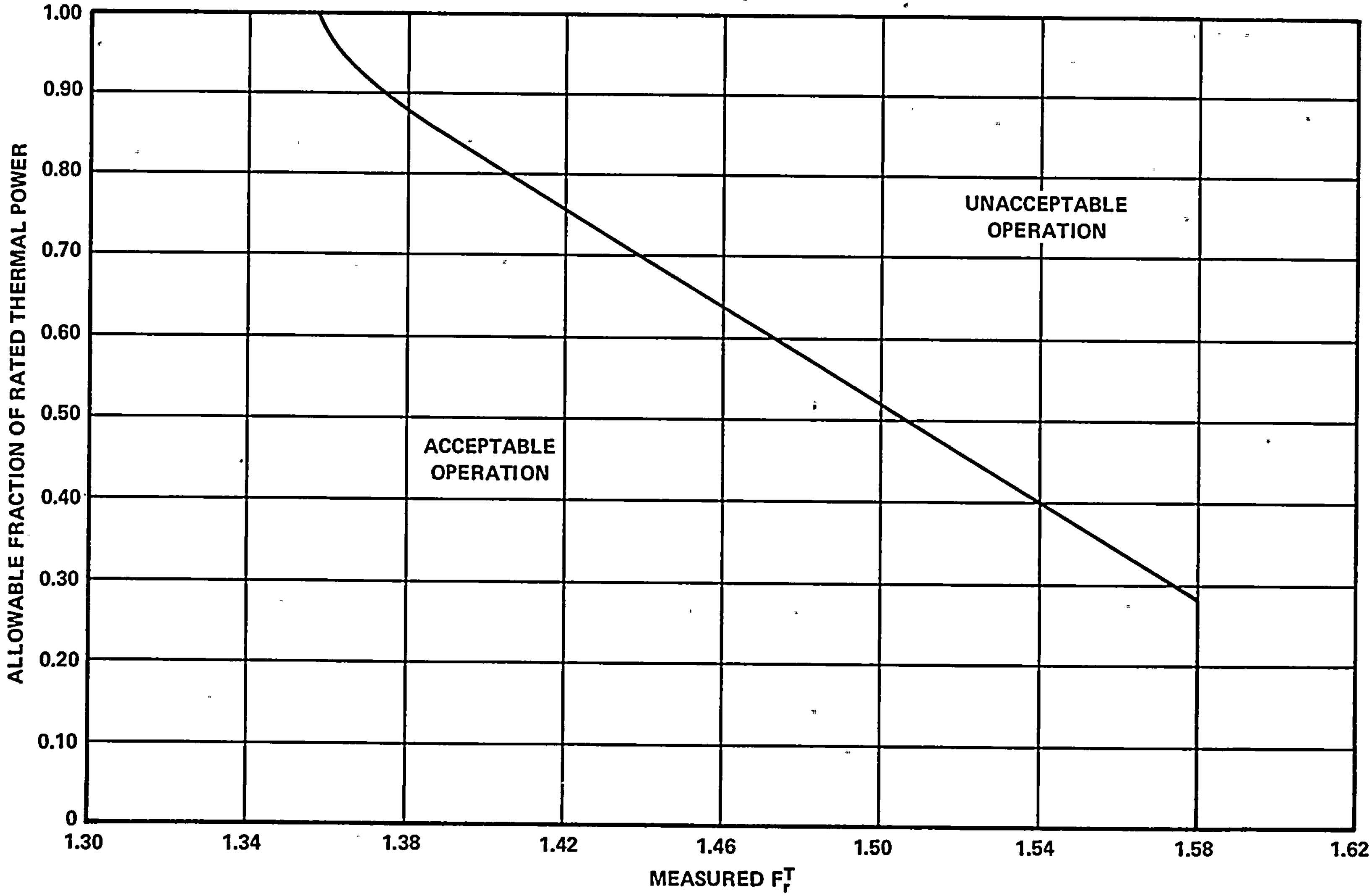


FIGURE 3.2-3

Allowable Combinations of THERMAL POWER
and F_r^T

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.3 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.02.

APPLICABILITY: MODES 1* and 2*.

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > 0.02 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL RADIAL PEAKING FACTOR (F_r^T) is within the limit of Specification 3.2.2.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided F_r^T or the combination of F_r^T and THERMAL POWER is maintained within the limit of Specification 3.2.2. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided:

   1. The THERMAL POWER level is restricted to $\leq 20\%$ of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination, and

   2. The Local Power Density-High and Thermal Margin/Low Pressure trip setpoints and Power Ratio Calculator setpoints are reduced by a factor equivalent to $\geq \frac{F_r^T \text{ (meas)}}{(1.36)}$.

SURVEILLANCE REQUIREMENT

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

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

a. Calculating the tilt at least once per 7 days when the Subchannel Deviation Alarm is OPERABLE,

---

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Calculating the tilt at least once per 12 hours when the Subchannel Deviation Alarm is inoperable, and

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

FUEL RESIDENCE TIME

LIMITING CONDITION FOR OPERATION

3.2.4 The core average fuel burnup shall be limited to $\leq$ 500 Effective Full Power Days during the initial fuel cycle.

APPLICABILITY: MODE 1.

ACTION:

With the core average fuel burnup determined to exceed 500 Effective Full Power Days, be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 The core average fuel burnup, based on gross thermal energy generation, shall be determined by calculation at least once per 31 days.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Cold Leg Temperature

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

d. AXIAL SHAPE INDEX

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 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 3.2-1

DNB MARGIN

LIMITS

Parameter	Four Reactor Coolant Pumps Operating
Cold Leg Temperature	≤ 542°F
Pressurizer Pressure	≥ 2225 psia*
Reactor Coolant Flow Rate	≥ 370,000 gpm
AXIAL SHAPE INDEX	Figure 3.2-4

*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.

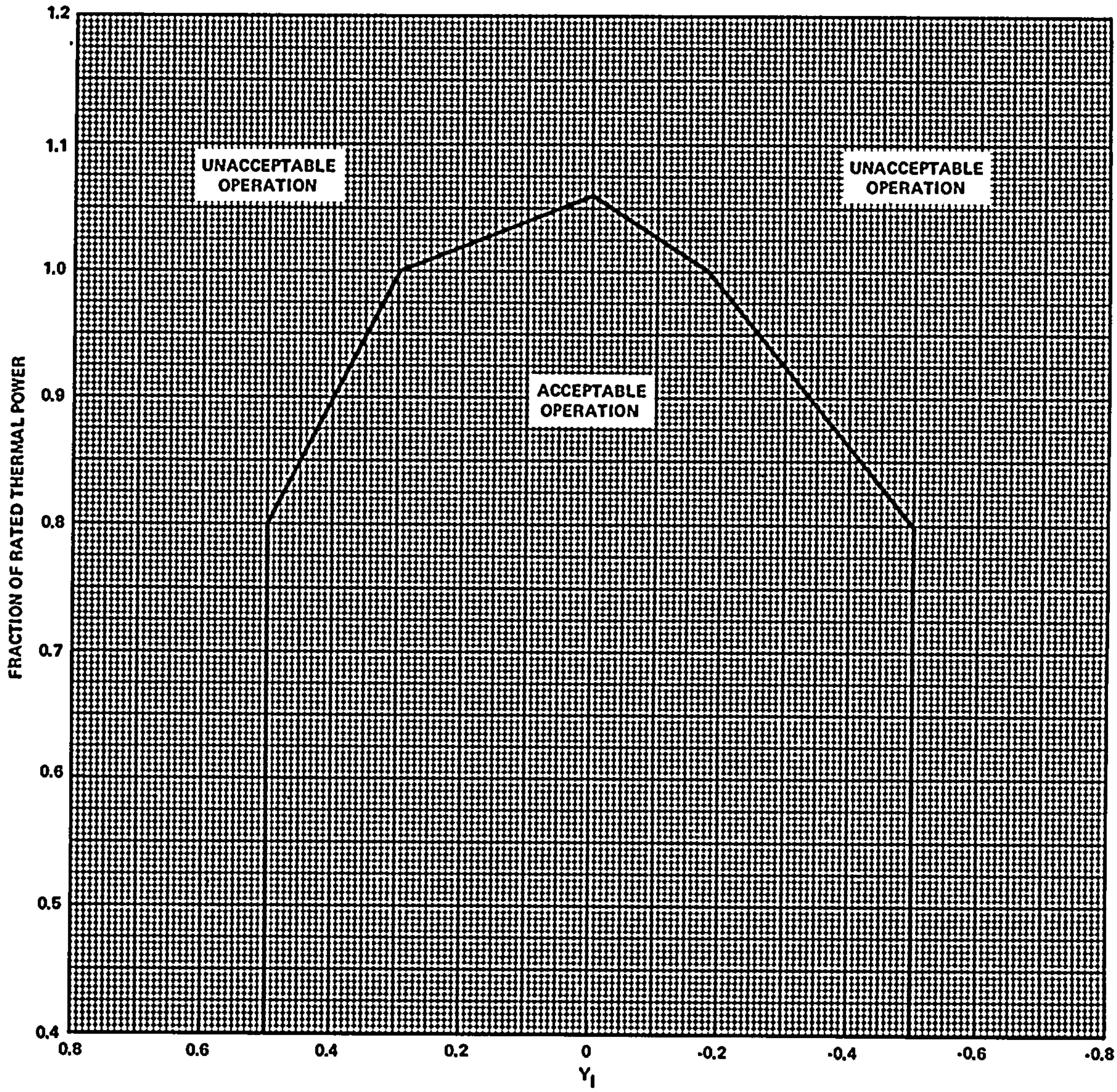


FIGURE 3.2-4
AXIAL SHAPE INDEX Operating Limits with 4 Reactor Coolant Pumps Operating

3/4.3.1 REACTOR PROTECTIVE INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 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

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

4.3.1.1.2 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.

4.3.1.1.3 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	2	1	2	1, 2 and *	1
2. Power Level - High	4	2(a)	3(f)	1, 2	2
3. Reactor Coolant Flow - Low	4/SG	2(a)/SG	3/SG	1, 2 (e)	2
4. Pressurizer Pressure - High	4	2	3	1, 2	2
5. Containment Pressure - High	4	2	3	1, 2	2
6. Steam Generator Pressure - Low	4/SG	2(b)/SG	3/SG	1, 2	2
7. Steam Generator Water Level - Low	4/SG	2/SG	3/SG	1, 2	2
8. Local Power Density - High	4	2(c)	3	1	2
9. Thermal Margin/Low Pressure	4	2(a)	3	1, 2 (e)	2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	4	2(c)	3	1	3

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating-- Rate of Change of Power - High	4	2(d)	3	1, 2 and *	3
b. Shutdown	4	0	2	3, 4, 5	4
12. Reactor Protection System Logic	2	1	2	1, 2*	5
13. Reactor Trip Breakers	2	1	2	1, 2*	5

TABLE NOTATION

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

(a) Trip may be bypassed below 5% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 5% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 585 psig; bypass shall be automatically removed at or above 585 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is ≥ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

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

(f) 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 HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels and with the THERMAL POWER level:

a. ≤ 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the following 6 hours.

b. > 5% of RATED THERMAL POWER, operation may continue provided all of the following conditions are satisfied:

TABLE 3.3-1 (Continued)

ACTION STATEMENTS

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

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

3. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.1.1 provided one of the inoperable channels is placed in the tripped condition.

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

a. $\leq$ 5% of RATED THERMAL POWER, place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing THERMAL POWER above 5% of RATED THERMAL POWER; otherwise, reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the following 6 hours.

b. > 5% of RATED THERMAL POWER, POWER OPERATION may continue provided all of the following conditions are satisfied:

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

   2. The Minimum Channels OPERABLE requirements is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.1.1 provided one of the inoperable channels is placed in the tripped condition.

ACTION 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 or 3.1.1.2, as applicable, within 1 hour and at least once per 12 hours thereafter.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours.

TABLE 3.3-2

REACTOR PROTECTIVE INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
1. Manual Reactor Trip	Not Applicable
2. Power Level - High	$\leq$ 0.40 seconds*
3. Reactor Coolant Flow - Low	$\leq$ 0.65 seconds
4. Pressurizer Pressure - High	$\leq$ 0.90 seconds
5. Containment Pressure - High	$\leq$ 1.40 seconds
6. Steam Generator Pressure - Low	$\leq$ 0.90 seconds
7. Steam Generator Water Level - Low	$\leq$ 0.90 seconds
8. Local Power Density - High	$\leq$ 0.40 seconds*
9. Thermal Margin/Low Pressure	$\leq$ 0.90 seconds*
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	Not Applicable
11. Wide Range Logarithmic Neutron Flux Monitor	Not Applicable

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

TABLE 4.3-1

REACTOR PROTECTIVE INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2. Power Level - High				
a. Nuclear Power	S	D(2), M(3),Q(5)	M	1, 2
b. ΔT Power	S	D(4), Q	M	1
3. Reactor Coolant Flow - Low	S	R	M	1, 2
4. Pressurizer Pressure - High	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 Water Level - Low	S	R	M	1, 2
8. Local Power Density - High	S	R	M	1
9. Thermal Margin/Low Pressure	S	R	M	1, 2
10. Loss of Turbine--Hydraulic Fluid Pressure - Low	N.A.	N.A.	S/U(1)	N.A.
11. Wide Range Logarithmic Neutron Flux Monitor	S	N.A.	S/U(1)	1, 2, 3, 4, 5 and *
12. Reactor Protection System Logic	N.A.	N.A.	M and S/U(1)	1, 2
13. Reactor Trip Breakers	N.A.	N.A.	M	1, 2 and *

TABLE 4.3-1 (Continued)

TABLE NOTATION

* - With reactor trip breaker closed.

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

(2) - Heat balance only, above 15% of RATED THERMAL POWER; adjust "Nuclear Power Calibrate" potentiometer to null "Nuclear Pwr - ΔT Pwr." During PHYSICS TESTS, these daily calibrations of nuclear power and ΔT power 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 RATER 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) - Adjust "ΔT Pwr Calibrate" potentiometers to make ΔT power signals agree with calorimetric calculation.

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

3/4.3.2 ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

4.3.2.1.2 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.

4.3.2.1.3 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 ESF function as shown in the "Total No. of Channels" Column of Table 3.3-3.

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION (SIAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure - High	4	2	3	1, 2, 3	9
c. Pressurizer Pressure - Low	4	2	3	1, 2(d), 3(a)	9
2. CONTAINMENT SPRAY (CSAS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure -- High - High	4	2(b)	3	1, 2, 3	10
3. CONTAINMENT ISOLATION (CIS)					
a. Manual (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Containment Pressure - High	4	2	3	1, 2, 3	9
c. Containment Radiation - High	4	2	3	1, 2, 3, 4	9
4. MAIN STEAM LINE ISOLATION (MSIS)					
a. Manual (Trip Buttons)	2/steam generator	1/steam generator	2/operating steam generator	1, 2, 3, 4	8
b. Steam Generator Pressure - Low	4/steam generator	2/steam generator	3/steam generator	1, 2, 3(c)	9

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	9
6. LOSS OF POWER					
4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	1/Bus	1/Bus	1/Bus	1, 2, 3	9

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1725 psia.

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

(c) Trip function may be bypassed in this MODE below 585 psig; bypass shall be automatically removed at or above 585 psig.

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

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels and with the pressurizer pressure:

a. < 1725 psia; place the inoperable channel in the tripped condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing the pressurizer pressure above 1725 psia; otherwise, be in at least HOT STANDBY within the following 6 hours.

b. ≥ 1725 psia, operation may continue provided all of the following conditions are satisfied:

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

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

3. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be removed from service for up to 2 hours for surveillance testing per Specification 4.3.2.1 provided one of the inoperable channels is placed in the tripped condition.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels and with the pressurizer pressure:

a. < 1725 psia; place the inoperable channel in the bypassed condition within 1 hour and restore the inoperable channel to OPERABLE status within 24 hours after increasing the pressurizer pressure above 1725 psia; otherwise, be in at least HOT SHUTDOWN within the following 12 hours.

b. ≥ 1725 psia, demonstrate that the Minimum Channels OPERABLE requirement is met within 1 hour; operation may continue with the inoperable channel bypassed and 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 FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 5 psig	$\leq$ 5 psig
c. Pressurizer Pressure - Low	$\geq$ 1600 psia	$\geq$ 1600 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	$\leq$ 10 psig	$\leq$ 10 psig
3. CONTAINMENT ISOLATION (CIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	$\leq$ 5 psig	$\leq$ 5 psig
c. Containment Radiation - High	$\leq$ 10 R/hr	$\leq$ 10 R/hr
4. MAIN STEAM LINE ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	$\geq$ 485 psig	$\geq$ 485 psig
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	48 inches above tank bottom	48 inches above tank bottom

TABLE 3.3-4 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP VALUE	ALLOWABLE VALUES
6. LOSS OF POWER		
4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	$\geq$ 3307 volts	$\geq$ 3307 volts

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 Fan Coolers	Not Applicable
Feedwater Isolation	Not Applicable
b. CSAS	
Containment Spray	Not Applicable
c. CIS	
Containment Isolation	Not Applicable
Shield Building Ventilation System	Not Applicable
d. RAS	
Containment Sump Recirculation	Not Applicable
e. MSIS	
Main Steam Isolation	Not Applicable
2. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	$\leq$ 30.0*/19.5**
b. Containment Fan Coolers	$\leq$ 30.0*/17.0**
c. Feedwater Isolation	$\leq$ 60.0
3. Containment Pressure-High	
a. Safety Injection (ECCS)	$\leq$ 30.0*/19.5**
b. Containment Isolation	$\leq$ 30.5*/20.5**
c. Shield Building Ventilation System	$\leq$ 30.0*/14.0**
d. Containment Fan Coolers	$\leq$ 30.0*/17.0**
e. Feedwater Isolation	$\leq$ 60.0

TABLE 3.3-5 (Continued)

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
4. Containment Pressure--High-High	
a. Containment Spray	$\leq$ 30.0*/18.5**
5. Containment Radiation-High	
a. Containment Isolation	$\leq$ 30.5*/20.5**
b. Shield Building Ventilation System	$\leq$ 30.0*/14.0**
6. Steam Generator Pressure-Low	
a. Main Steam Isolation	$\leq$ 6.9
7. Refueling Water Storage Tank-Low	
a. Containment Sump Recirculation	$\leq$ 91.5
8. Reactor Trip	
a. Feedwater Flow Reduction to 5%	$\leq$ 60.0

TABLE NOTATION

*Diesel generator starting and sequence loading delays included.

**Diesel generator starting and sequence loading delays not included. Offsite power available.

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
6. LOSS OF POWER 4.16 kv Emergency Bus Undervoltage (Undervoltage relays)	S	R	M	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) The logic circuits shall be tested manually at least once per 31 days.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING

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 setpoints within the specified limits.

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm 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 during 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 SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area	1	*	$\leq$ 15 mR/hr	$10^{-1} - 10^{4}$ mR/hr	13
b. Containment (CIS)	3	6	$\leq$ 90 mR/hr	$1 - 10^{5}$ mR/hr	16
2. PROCESS MONITORS					
a. Containment					
i. Gaseous Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm	14
ii. Particulate Activity RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	$10 - 10^{6}$ cpm/hr	14

* With fuel in the storage pool or building

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 13 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, perform area surveys of the monitored area with portable monitoring instrumentation at least once per 24 hours.

ACTION 14 - 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 16 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, comply with the ACTION requirements of Specification 3.9.9.

TABLE 4.3-3

RADIATION MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. AREA MONITORS				
a. Fuel Storage Pool Area	S	R	M	*
b. Containment (CIS)	S	R	M	6
2. PROCESS MONITORS				
b. Containment				
i. Gaseous Activity RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Particulate Activity RCS Leakage Detection	S	R	M	1, 2, 3, & 4

* With fuel in the storage pool or building

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.

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:

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

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detection system shall be demonstrated OPERABLE:

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

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 channels 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 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 system to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

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

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT CHANNEL	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	N.A.	R	N.A.
b. SMR-42-7	N.A.	R	N.A.
c. SMR-42-8	N.A.	R	N.A.
3. PEAK SHOCK RECORDERS			
a. SMR-42-9	N.A.	R	N.A.
b. SMR-42-10	N.A.	R	N.A.
4. EARTHQUAKE FORCE MONITOR			
a. SMI-42-11	M	R	SA
5. SEISMIC SWITCH			
a. SMS-42-12	N.A.	R	SA

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

Table 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	INSTRUMENT MINIMUM ACCURACY	MINIMUM CHANNELS OPERABLE
1. WIND SPEED			
	a. Nominal Elev. 32.8 ft.	± 0.5 mph*	1
	b. Nominal Elev. 190 ft.	± 0.5 mph*	1
2. WIND DIRECTION			
	a. Nominal Elev. 32.8 ft.	± 5°	1
	b. Nominal Elev. 190 ft.	± 5°	1
3. AIR TEMPERATURE - DELTA T			
	a. Nominal Elev. 32.8 ft.	± 0.18°F	1
	b. Nominal Elev. 110 ft.	± 0.18°F	1
	c. Nominal Elev. 200 ft.	± 0.18°F	1

*Starting speed of anemometer shall be < 1 mph.

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 32.8 ft.	D	SA
b. Nominal Elev. 190 ft.	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 32.8 ft.	D	SA
b. Nominal Elev. 190 ft.	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 32.8 ft.	D	SA
b. Nominal Elev. 110 ft.	D	SA
c. Nominal Elev. 200 ft.	D	SA

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the number of OPERABLE remote shutdown monitoring channels less than required by Table 3.3-9, either:

a. Restore the inoperable channel to OPERABLE status within 30 days, or

b. Be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Reactor Trip Breaker Indication	SWGR	OPEN-CLOSE	1/trip breaker
2. Pressurizer Pressure	Hot Shutdown Panel	1500-2500 psia	1
3. Pressurizer Level	Hot Shutdown Panel	0-100%	1
4. Main Steam Pressure	Hot Shutdown Panel	0-1200 psig	1/steam generator
5. Steam Generator Level	Hot Shutdown Panel	0-100%	1/steam generator
6. Cold Leg Temperature	Hot Shutdown Panel	0-600°F	1

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N.A.
2. Pressurizer Pressure	M	R
3. Pressurizer Level	M	R
4. Steam Generator Level	M	R
5. Main Steam Pressure	M	R
6. Cold Leg Temperature	M	R

CHLORINE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.3.3.6 Two separate and independent chlorine detection systems, with their alarm/trip setpoints adjusted to actuate at a chlorine concentration of $\leq$ 5 ppm, shall be OPERABLE with each chlorine detection system having at least one chlorine detector in each control room outside air intake duct.

APPLICABILITY: ALL MODES

ACTION:

a. With one chlorine detector inoperable, within 2 hours either isolate the associated outside air intake or initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation until the inoperable chlorine detector is restored to OPERABLE status.

b. With no chlorine detection system OPERABLE, initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation until two chlorine detection systems are restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.3.3.6 Each chlorine detection system shall be verified energized at least once per 12 hours and demonstrated OPERABLE by performance of a CHANNEL CALIBRATION at least once per 18 months.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Four reactor coolant pumps shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.*

ACTION:

MODES 1 and 2:

With less than four reactor coolant pumps in operation, be in at least HOT STANDBY within 6 hours.

MODES 3, 4 and 5:

Operation may proceed provided at least once reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump; however, operation for up to 15 minutes with no pump in operation is permissible to accommodate transition between shutdown cooling pump and reactor coolant pump operation. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1 The Flow Dependent Selector Switch shall be determined to be in the 4 pump position within 15 minutes prior to making the reactor critical and at least once per 12 hours thereafter.

* See Special Test Exception 3.10.4.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 4 and 5.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Reguirement 4.4.3.

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.3 Each pressurizer code safety valve shall be demonstrated OPERABLE with a lift setting of 2500 PSIA ± 1%, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with a steam bubble.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.4 Not applicable.

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

4.4.5.1 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. Steam generator tubes shall be examined in accordance with Appendix IV of the ASME Boiler and Pressure Vessel Code - Section XI - "Inservice Inspection of Nuclear Power Plant Components" 1974 Edition and Addenda through Summer 1976. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all steam generators; the tubes selected for these inspections shall be selected on a random basis except:

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

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

    1. All nonplugged tubes that previously had detectable wall penetrations (>20%), and

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

c. The second and third inservice inspections may be less than a full tube inspection by concentrating (selecting at least 50% of the tubes to the inspected) the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with 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 (>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 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the inservice inspection of a steam generator conducted in accordance with Table 4.4-2 requires a third sample inspection whose results fall in Category C-3, the inspection frequency shall be reduced to at least once per 20 months. The reduction in inspection frequency shall apply until a subsequent inspection demonstrates that a third sample inspection is not required.

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 safeguards, or

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 $\geq$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. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

6. Plugging Limit means the imperfection depth at or beyond which the tube shall be removed from service because it may become unserviceable prior to the next inspection and is equal to 40% of the nominal tube wall thickness.

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

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

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

b. The complete results of the steam generator tube inservice inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S. G.</td><td>C–1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C–2</td><td rowspan="5">Plug defective tubes and inspect additional 2S tubes in this S. G.</td><td>C–1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C–2</td><td rowspan="3">Plug defective tubes and inspect additional 4S tubes in this S. G.</td><td>C–1</td><td>None</td></tr>
<tr><td>C–2</td><td>Plug defective tubes</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td></tr>
<tr><td>C–3</td><td>Perform action for C–3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C–3</td><td rowspan="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</td><td>All other S. G.s are C–1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some S. G.s C–2 but no additional S. G. are C–3</td><td>Perform action for C–2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional S. G. is C–3</td><td>Inspect all tubes in each S. G. and plug defective tubes. Prompt notification to NRC pursuant to specification 6.9.1</td><td>N/A</td><td>N/A</td></tr>
</table>

$S = 3\frac{N}{n}\%$ 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:

a. With one of the above required radioactivity monitoring leakage detection systems inoperable, operations may continue for up to 30 days provided:

   1. The other two above required leakage detection systems are OPERABLE, and

   2. Appropriate grab samples are obtained and analyzed at least once per 24 hours;

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

b. With both the above required radioactivity monitoring leakage detection systems inoperable, operations may continue for up to 30 days provided:

   1. The reactor cavity sump level and flow monitoring system is OPERABLE,

   2. Appropriate grab samples are obtained and analyzed at least once per 24 hours, and

   3. A Reactor Coolant System water inventory balance is performed at least once per 8 hours during steady state operation except when operating in the shutdown cooling mode;

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

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

c. With the containment sump level and flow monitoring 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

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

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

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

REACTOR COOLANT SYSTEM 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 steam generators, and

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.4.6.2 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 at least once per 12 hours.

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

c. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation except when operating in the shutdown cooling mode, and

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

LIMITING CONDITION FOR OPERATION

3.4.7 The Reactor Coolant System chemistry shall be maintained within the limits specified in Table 3.4-1.

APPLICABILITY: ALL MODES.

ACTION:

MODES 1, 2, 3 and 4

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

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

MODES 5 and 6

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the pressurizer pressure to $\leq$ 500 psia, if applicable, and perform an analysis 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 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 3.4-1

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS

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

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

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 $T_{avg} \leq 250°F$.

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

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

a. $\leq$ 1.0 μCi/gram DOSE EQUIVALENT I-131, and

b. $\leq$ 100/$\bar{E}$ μCi/gram.

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

ACTION:

MODES 1, 2 and 3*:

a. With the specific activity of the primary coolant > 1.0 μCi/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 100 hours provided that operation under these circumstances shall not exceed 10 percent of the unit's total yearly operating time. The provisions of Specification 3.0.4 are not applicable.

b. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 for more than 100 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with T_{avg} < 500°F within 6 hours.

c. With the specific activity of the primary coolant > 100/$\bar{E}$ μCi/gram, be in HOT STANDBY with T_{avg} < 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5:

d. With the specific activity of the primary coolant > 1.0 μCi/gram DOSE EQUIVALENT I-131 or > 100/$\bar{E}$ μCi/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,

*With $T_{avg} \geq$ 500°F.

ACTION: (Continued)

2. Fuel burnup by core region,

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

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

5. The time duration when the specific activity of the primary coolant exceeded 1.0 μCi/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	MINIMUM FREQUENCY	MODES IN WHICH SAMPLE AND ANALYSIS REQUIRED
1. Gross Activity Determination	3 times per 7 days with a maximum time of 72 hours between samples	1, 2, 3 and 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 DOSE EQUIVALENT I-131 exceeds 1.0 µCi/gram, and	1#, 2#, 3#, 4# and 5#
	b) One sample between 2 and 6 hours following a THERMAL POWER change exceeding 15 percent of the RATED THERMAL POWER within a one hour period.	1, 2, 3

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

*After at least 2 EFPD and at least 20 days since the last shutdown of longer than 48 hours.

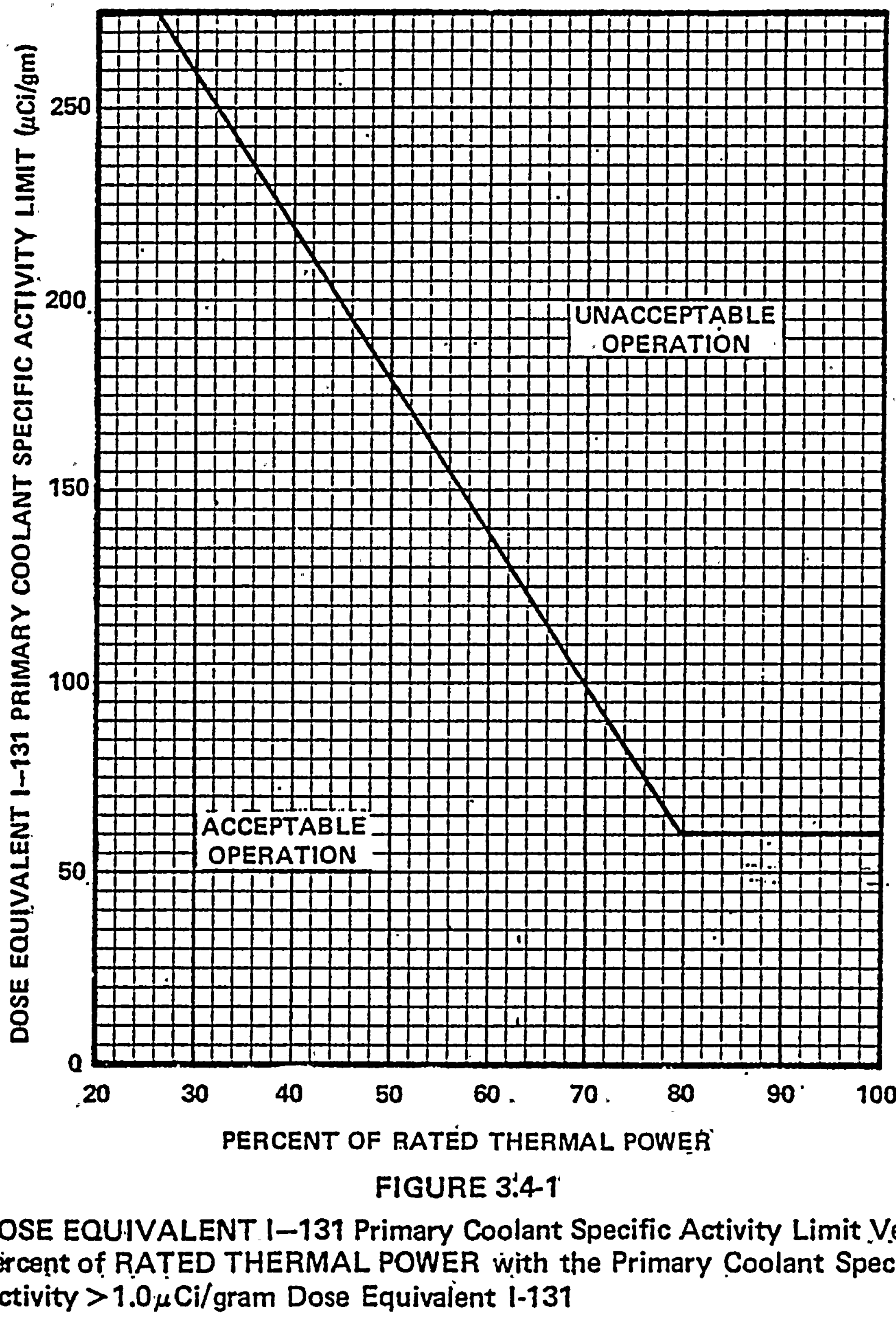


FIGURE 3.4-1

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

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

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

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

c. A maximum temperature change of 5°F in any one hour period, during hydrostatic testing operations above system design pressure.

APPLICABILITY: At all times.*

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued 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.

*See Special Test Exception 3.10.3.

SURVEILLANCE REQUIREMENTS

4.4.9.1

a. 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.

b. The Reactor Coolant System temperature and pressure conditions shall be determined to be to the right of the criticality limit line within 15 minutes prior to achieving reactor criticality.

c. The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figure 3.4-2.

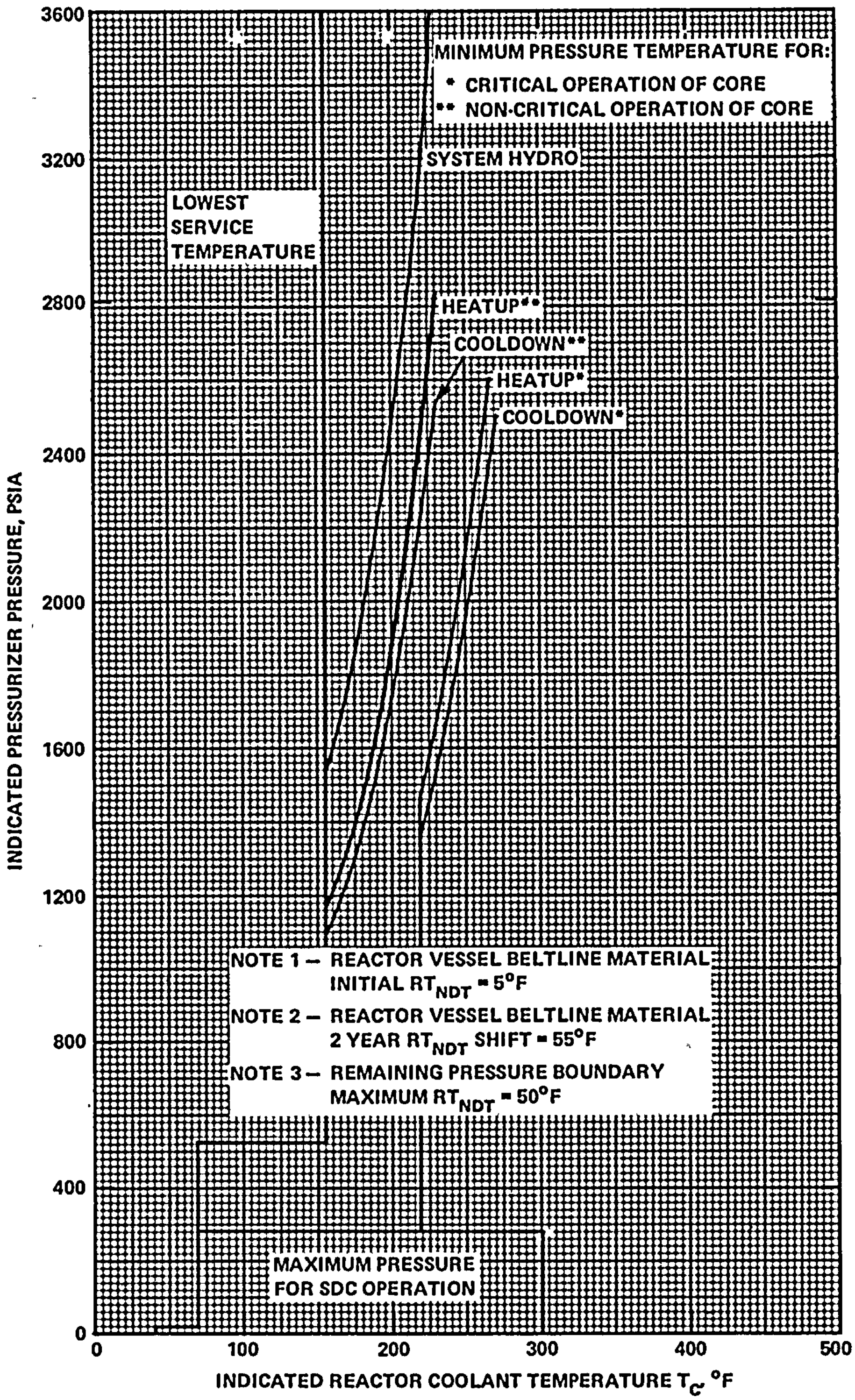


FIGURE 3.4-2a

Reactor Coolant System Pressure Temperature Limitations for 0 to 2 Years of Full Power Operation

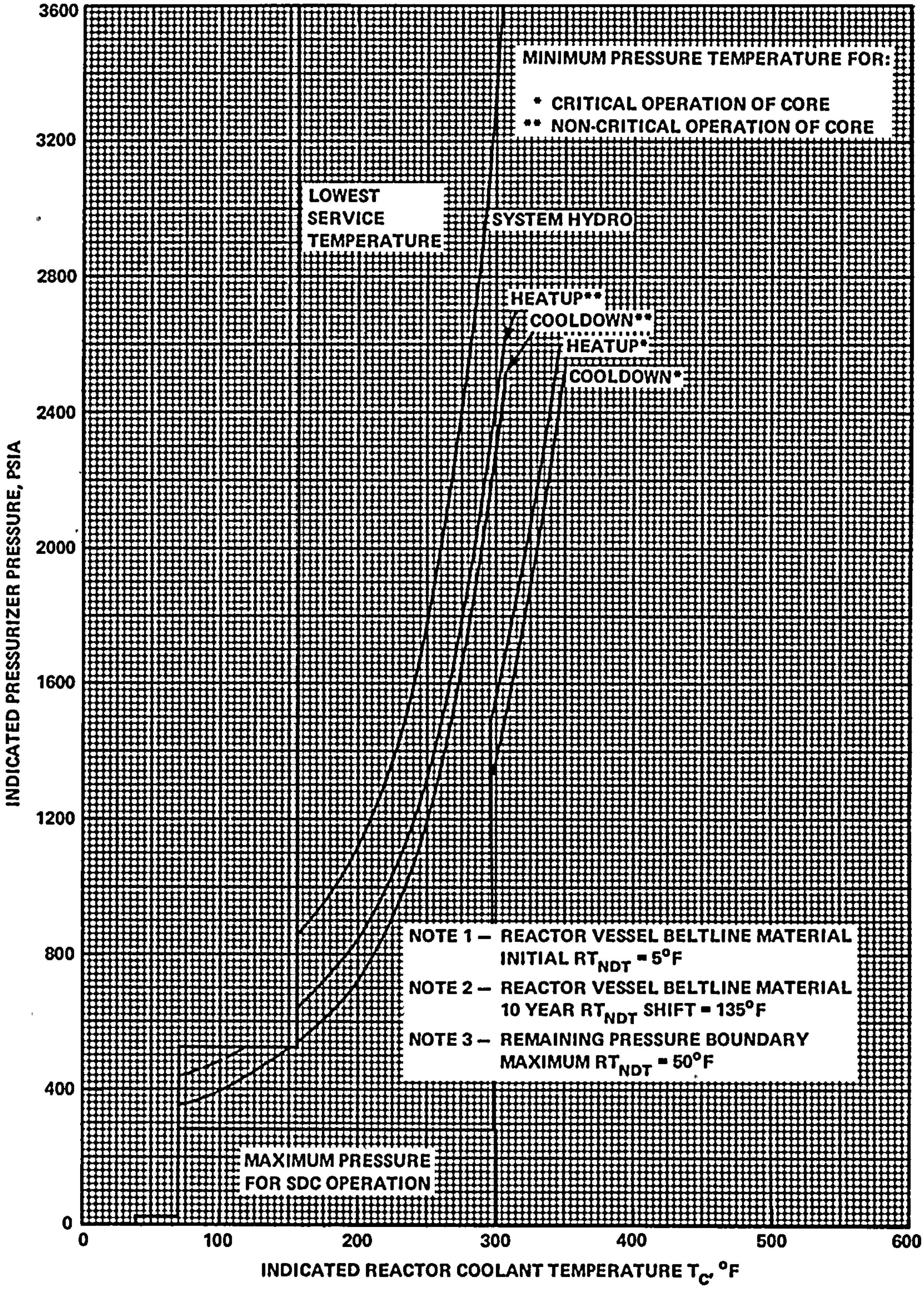


FIGURE 3.4-2b

Reactor Coolant System Pressure Temperature Limitations for 2 to 10 Years of Full Power Operation

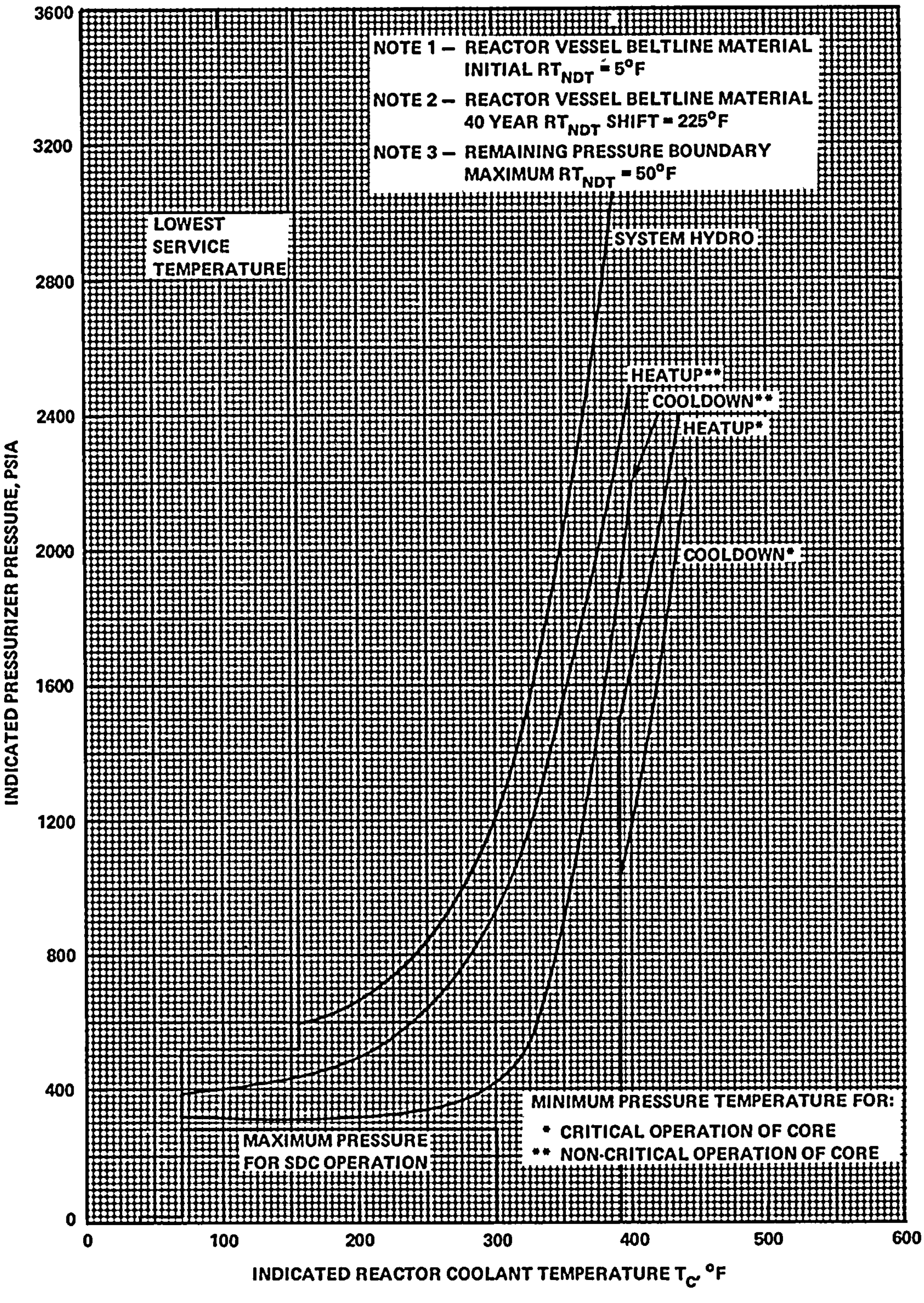


FIGURE 3.4-2c

Reactor Coolant System Pressure Temperature Limitations for 10 to 40 Years of Full Power Operation

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

SPECIMEN	REMOVAL INTERVAL
1.	8 years
2.	16 years
3.	23 years
4.	30 years
5.	35 years
6.	40 years

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

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

c. A maximum Reactor Coolant System spray water temperature differential of 350°F.

APPLICABILITY: At all times.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

SAFETY CLASS 1 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of components (except steam generator tubes) identified in Section 3.2.2 of the FSAR as Safety Class 1 components shall be maintained at a level consistent with the acceptance criteria in Specification 4.4.10.1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of any of the above components not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolated the affected component prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1 The following inspection program shall be performed during shutdown:

a. Inservice Inspections The structural integrity of the Safety Class 1 components shall be demonstrated by verifying their acceptability when inspected per the applicable requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and Addenda through Winter 1972, as outlined by the inspection program shown in Table 4.4-6.

   An initial report of any abnormal degradation of the structural integrity of the Safety Class 1 components detected during the above required inspections shall be made within 10 days after detection and the detailed report shall be submitted pursuant to Specification 6.9.1 within 90 days after completion of the surveillance requirements of this specification.

SURVEILLANCE REQUIREMENTS (Continued)

The Inservice Inspection Program shall be reviewed every 5 years to assure that the equipment, techniques and procedures being utilized are current and applicable. The results of these reviews shall be reported in Special Reports to the Commission pursuant to Specification 6.9.2 within 90 days of completion.

b. Inspections Following Repairs or Replacements The structural integrity of the reactor coolant system shall be demonstrated after completion of all repairs and/or replacements to the system by verifying the repairs and/or replacements meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and Addenda through Winter 1972. When repairs and/or replacements are made which involve new strength welds on components greater than 2 inch diameter, the new welds shall receive a surface and 100 percent volumetric examination and meet applicable code requirements. When repairs and/or replacements are made which involve new strength welds on components 2 inch diameter or smaller, the new welds shall receive a surface examination and meet applicable code requirements.

c. Inspections Following System Opening The structural integrity of the reactor coolant system shall be demonstrated after each closing by performing a leak test, with the system pressurized to at least 2235 psig, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and Addenda through Winter 1972, and the Pressure/Temperature limits of Specification 3.4.9.1.

TABLE 4.4-6

INSERVICE INSPECTION PROGRAM - SAFETY CLASS 1 COMPONENTS

SECTION 1. REACTOR VESSEL AND CLOSURE HEAD

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
1.1	A	Longitudinal and Circumferential shell welds in core region.	Volumetric	See Note (a) and Remarks	The required examinations will be performed at or near the end of the 10-year inspection interval or when the internals are removed for other reasons. Mechanized Ultrasonic examination will be used.
1.2	B	Longitudinal and circumferential welds in shell (other than those of Category A and C).	Volumetric	See Note (a) and Remarks	(1) The required amount of weld lengths will be examined at or near the end of the 10-year inspection interval or when the internals are removed for other reasons. (2) Closure head meridional welds and dome (circumferential) welds which are not accessible inside the head cooling shroud assembly will be excluded from the examination. (3) A partial examination of the RPV lower head welds can be done from the ID with mechanized techniques. a) Approximately 25% of meridional welds. b) Approximately 75% of circumferential weld (lower head to shell). c) 100% of the dome welds (circumferential). (4) 100% manual examination of RPV lower head welds will be performed externally during baseline.

TABLE 4.4-6 (Cont'd)

SECTION 1. REACTOR VESSEL AND CLOSURE HEAD

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
1.3	C	Vessel-to-flange and head-to-flange circumferential welds.	Volumetric	See Note (a) and Remarks	Both of these welds are available for examination during normal refueling operations.
1.4	D	Primary nozzle-to-vessel welds and nozzle-to-vessel inside radiused sections.	Volumetric	See Note (a) and Remarks	The required examinations will be performed by mechanized techniques as follows: (1) Whenever the internals are removed, one (1) outlet and one (1) inlet nozzle during the first two thirds of this interval (not to exceed 2 each). Balance per remark (2) below. (2) In case the internals are not removed, two (2) outlet nozzles for the first two thirds of the interval and four (4) inlet nozzles near or at the end of the interval.
1.5	E-1	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds.	Volumetric	See Remarks	(1) The CRDM tubes and in-core instrumentation tubes are welded to the upper head with a partial penetration weld. The assemblies contain an integrally welded thermal sleeve Volumetric examination is not feasible by currently available techniques. These penetrations are included in Category E-2. (2) Housing pressure boundary welds are inaccessible due to head cooling shroud assembly. (3) Meets exclusion criteria of paragraph IS-121(a) ref: FSAR 15.4.5.
1.6	E-2	Vessel penetrations, including control rod drive penetrations and control rod housing pressure boundary welds.	Visual	See Note (a)	

TABLE 4.4-6 (Cont'd)

SECTION 1. REACTOR VESSEL AND CLOSURE HEAD

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
1.7	F	Primary nozzles to safe end welds.	Visual, surface and volumetric.	See Remarks	There are no welds in this category.
1.8	G-1	Closure studs and nuts.	Visual, surface and volumetric	See Note (a) and Remarks	Surface examination does not apply to nuts.
1.9	G-1	Ligaments between threaded stud holes.	Volumetric	See Note (a) and Remarks	The ligaments will be examined whenever the Reactor Vessel Flange Weld of Item 1.3 is examined.
1.10	G-1	Closure washers, bushings.	Visual	See Note (a)	There are no bushings.
1.11	G-2	Pressure retaining bolting.	Visual	See Remarks	There is no bolting less than 2-inches in diameter accessible for examination.
1.12	H	Integrally welded vessel supports.	Volumetric	See Remarks	Nozzle-type pad supports are installed and are examined under Item 1.4, Category D.
1.13	I-1	Closure head cladding	Visual and surface or volumetric	See Note (a) and Remarks	At least six (6) patches (each 36 sq. in.) of the RPV Closure Head will be examined - visual and surface.
1.14	I-1	Vessel cladding.	Visual	See Note (a) and Remarks	At least six patches (each 36 sq. in.) of the vessel will be examined by remote techniques.
1.15	N	Interior surfaces and internals and integrally welded internal supports.	Visual	See Note (a) and Remarks	(1) The examinations will be done by remote visual techniques. (2) Whenever refueling occurs and internals are removed, a remote visual examination will be performed on the vessel interior.

TABLE 4.4-6 (Cont'd)

SECTION 2. PRESSURIZER

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
2.1	B	Longitudinal and Circumferential Welds	Visual and Volumetric	See Note (a)	
2.2	D	Nozzle-to-Vessel Welds and nozzle-to-vessel inside radius section	Volumetric	See Note (a)	
2.3	E-1	Heater Connections	Visual and Surface	See remarks	(1) Meet the exclusion criteria of IS-121 (c) and are examined under category E-2.
2.4	E-2	Heater Connections	Visual	See Note (a)	
2.5	G-1	Pressure Retaining Bolting	Visual and Volumetric	See Remarks	(1) Bolting 2-inch in diameter and greater do no exist.
2.6	G-2	Pressure retaining Bolting	Visual	See Note (a) and Remarks	The bolting less that 2-inches in diameter will be visually inspected, either in-place under tension or when they are disassembled.

TABLE 4.4-6 (Cont'd)

SECTION 2. PRESSURIZER

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
2.7	H	Integrally welded vessel supports.	Visual and Volumetric	See Note (a)	
2.8	I-2	Vessel cladding.	Visual	See Note (a) and Remarks	At least one (1) patch (36 sp. in.) below the primary manway will be examined.

SECTION 3. STEAM GENERATORS

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
3.1	B	Longitudinal and circumferential welds, including tube sheet-to-head welds on the primary side.	Visual and volumetric	See Note (a) and Remarks	Mechanized examination, no visual, will be performed on three (3) each staywell circumferential welds due to limited access and undue radiation exposure to personnel.
3.2	D	Primary nozzle-to-head welds and nozzle-to-head inside radiused sections.	Volumetric	See Not (a)	
3.3	F	Primary nozzle-to-safe end welds.	Visual, surface and volumetric.	See Remarks	Not Applicable.
3.4	G-1	Pressure retaining bolting.	Visual and volumetric	See Remarks	Not Applicable.
3.5	G-2	Pressure retaining bolting.	Visual	See Note (a) and Remarks	The bolting less that 2-inches in diameter will be visually examined in place under tension, or whenever the bolting connection is disassembled.
3.6	H	Integrally welded vessel supports.	Visual and volumetric	See Note (a)	

TABLE 4.4-6 (Cont'd)

SECTION 3. STEAM GENERATORS

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
3.7	I-2	Vessel cladding	Visual	See Note (a) and Remarks	Radiation levels permitting, the examinations will be conducted.
			SECTION 4. PIPING PRESSURE BOUNDARY		
4.1	F	Vessel, pump, and valve Safe ends-to-primary pipe welds and safe ends in branch piping welds.	Visual, surface and volumetric	See Note (a) and Remarks	The pressurizer and pump nozzle-to-safe end welds are included in this item.
4.2	G-1	Pressure retaining bolting.	Visual and volumetric	See Remarks	There is no bolting 2-inches and larger in the piping system.
4.3	G-2	Pressure retaining bolting.	Visual	See Note (a) and Remarks	All bolting below 2-inches in diameter will be visually examined, either in-place or whenever the bolted connection is disassembled.
4.4	J-1	Circumferential and longitudinal pipe welds.	Visual and volumetric	See Note (a)	An augmented inspection of the following welds shall be performed: (1) During first inspection interval-100% of the longitudinal welds from the reactor vessel to approximately the outboard surfaces of the reactor cavity shall be examined at approximately 3 1/3 year intervals in each of the two hot legs of the unit. (2) During subsequent inspection intervals - provided no defects are found during the first inspection interval exceeding the allowable limits provided in Section XI of the ASME Code, 100% of these longitudinal welds in either one of the two hot legs shall be examined during each inspection interval following the first.

TABLE 4.4-6 (Cont'd)

SECTION 4. PIPING PRESSURE BOUNDARY

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
4.5	J-1	Branch pipe connection welds, exceeding 4-in. nominal pipe size.	Visual and volumetric	See Note (a)	
4.6	J-1	Socket Welds.	Visual and surface	See Note (a)	
4.7	J-1	Branch pipe connection welds, 4-in. nominal pipe size and smaller.	Visual and surface	See Note (a)	
4.8	J-2	Circumferential and longitudinal pipe welds and branch pipe connection welds.	Visual	See Note (a)	
4.9	K-1	Integrally welded supports.	Visual and volumetric	See Notes (a) and (d)	
4.10	K-2	Piping supports and hangers.	Visual	See Note (a)	
		SECTION 5. PUMP PRESSURE BOUNDARY AND PUMP FLYWHEELS			
5.1	L-1	Pump casing welds.	Visual and volumetric	See Note (a) and Remarks	The only feasible method known to date to volumetrically examine these pump casing welds is radiography. If radiography is possible, or an alternate examination method is developed, the examination will be performed at the frequency indicated. See Note (c).
5.2	L-2	Pump casings.	Visual	See Note (a) and Remarks	Radiation levels permitting, the examination will be performed whenever the pumps are disassembled.
5.3	F	Nozzle-to-safe end welds.	Visual and volumetric	See Note (a) and Remarks	This item is considered under Section 4, Item 4.1.

TABLE 4.4-6 (Cont'd)

SECTION 5. PUMP PRESSURE BOUNDARY AND PUMP FLYWHEELS

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
5.4	G-1	Pressure retaining bolting.	Visual and volumetric	See Note (a) and Remarks	(1) Bolting 2-inches and larger in diameter will be examined either in-place under tension, or when the bolting is removed or when the bolting connection is disassembled. (2) There is no feasible method of examining the flange ligaments of the pumps volumetrically at the present time. See note (c).
5.5	G-2	Pressure retaining bolting.	Visual	See Remarks	There is no bolting below 2-inches in diameter.
5.6	K-1	Integrally welded supports.	Visual and volumetric	See Note (a)	
5.7	K-2	Supports and hangers.	Visual	See Note (a)	
5.8	---	Flywheel	Volumetric (1)(2) Surface (2)	See Remarks	(1) An in-place ultrasonic volumetric examination of the areas of higher stress concentration at the bore and key way at approximately 3 year intervals, during the refueling or maintenance shutdown coinciding with the inservice inspection schedule. (2) A surface examination of all exposed surfaces and complete ultrasonic volumetric examination at approximately 10 year intervals, during the plant shutdown coinciding with the inservice inspection schedule. Removal of the flywheel is not required to perform these examinations.

TABLE 4.4-6 (Cont'd)

SECTION 6. VALVE PRESSURE BOUNDARY

Item No.(b)	Category(b)	Examination Area(b)	Examination Method(b)	Tentative Inspection During 10-year Interval	Remarks
6.1	M-1	Valve body welds	Visual and volumetric	Remarks	Not Applicable.
6.2	M-2	Valve bodies	Visual	See Note (a) and Remarks	Examinations will be conducted provided valves can be disassembled without undue radiation exposure to personnel.
6.3	F	Valve-to-safe end welds.	Visual and volumetric	See Remarks	Not Applicable.
6.4	G-1	Pressure retaining bolting.	Visual and volumetric	See Remarks	There are no valves with bolting 2-inches and larger in diameter.
6.5	G-2	Pressure retaining bolting.	Visual	See Note (a) and Remarks	All bolting below 2-inches in diameter will be visually examined, either in-place if the bolting connection is not disassembled during the inspection interval, or whenever the bolting connection is disassembled.
6.6	K-1	Integrally welded supports	Visual and volumetric	See Remarks	There are no valves with integrally welded supports.
6.7	K-2	Support and hangers	Visual	See Note (a)	

Note (a): The extent and frequency of the examinations will be at least as much as the guidelines of Table IS-251 of Section XI 1971 Edition through Winter 1972 Addenda.

(b): The item number, category, examination area and examination method are listed in Table IS-261 of Section XI of the ASME Boiler and Pressure Vessel Code.

(c): Present ultrasonic techniques may not be amenable to examination of pump/valve castings.

(d): Where accessibility to welds prohibits their examination, each weld shall be examined on a case by case basis.

SAFETY CLASS 2 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.2 The structural integrity of components identified in Section 3.2.2 of the FSAR as Safety Class 2 components shall be maintained at a level consistent with the acceptance criteria in Specification 4.4.10.2.

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

ACTION:

With the structural integrity of any of the above components not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component prior to increasing the Reactor Coolant System above 200°F. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.2 The following inspection program shall be performed during shutdown:

a. Inservice Inspections The structural integrity of the Safety Class 2 components shall be demonstrated by verifying their acceptability when inspected per the applicable requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and Addenda through Winter 1972, as outlined by the inspection program shown in Table 4.4-7.

   An initial report of any abnormal degradation of the structural integrity of the Safety Class 2 components detected during the above required inspections shall be made within 10 days after detection and the detailed report shall be submitted pursuant to Specification 6.9.1 within 90 days after completion of the surveillance requirements of this specification.

SURVEILLANCE REQUIREMENTS (Continued)

The Inservice Inspection Program shall be reviewed every 5 years to assure that the techniques and procedures being utilized are current and applicable. The results of these reviews shall be reported in Special Reports to the Commission pursuant to Specification 6.9.2 within 90 days of completion.

b. Inspections Following Repairs or Replacements The structural integrity of Safety Class 2 components shall be demonstrated after completion of all repairs and/or replacements by verifying the repairs and/or replacements meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and Addenda through Winter 1972. When repairs and/or replacements are made which involve new strength welds on components greater than 2 inch diameter, the new welds shall receive a surface and 100 percent volumetric examination and meet applicable code requirements. When repairs and/or replacements are made which involve new strength welds on components 2 inch diameter or smaller, the new welds shall receive a surface examination and meet applicable code requirements.

c. System Pressure Tests The structural integrity of Safety Class 2 components shall be demonstrated by performing system pressure tests per the requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and Addenda through Winter 1972.

TABLE 4.4-7

INSERVICE INSPECTION PROGRAM - SAFETY CLASS 2 COMPONENTS

SECTION C1. PRESSURE VESSELS

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C1.1	C-A	Circumferential shell and head welds	Volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) The areas of examination include shell and head circumferential welds which are gross structural discontinuities (see Subparagraph NE-3213.2 of Section III - Nuclear Power Plant Components Code) in vessels exceeding 4-inch nominal pipe size. (2) See Note (f). (3) See Note (b). (4) The steam generator staywell dome welds will not be examined due to their configuration, limited access and radiation exposure to personnel.
C1.2	C-B	Nozzle-to-vessel welds	Volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) See Note (b). (2) See Note (f).

TABLE 4.4-7 (Cont'd)

SECTION C1. PRESSURE VESSELS

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C1.3	C-C	Integrally-welded supports	Surface	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) See Note (b). (2) See Note (f).
C1.4	C-D	Pressure-retaining bolting	Visual and either surface or volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) Bolting exceeding 1-inch in diameter will be examined in-place in the bolted-up condition or whenever the bolted connection is disassembled. Flange ligaments between threaded stud holes will be included whenever the connection is disassemble, and feasible methods of examination have been developed. (2) A required examination consists of the following: (a) Visual examinations will include 100% of the bolts, studs, and nuts in the bolted joint. (b) Surface examinations will be performed on 10% of the bolting in each joint, but not less than 2 bolts or studs per joint whenever the connection is disassembled. (3) See Note (b). (4) See Note (f).

TABLE 4.4-7 (Cont'd)

SECTION C1. PRESSURE VESSELS

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C1.5	---	ISC-261(c) components	Visual	Cumulative 100% of the required examinations (see Note (c)).	(1) Examinations will be conducted in accordance with the procedures of IS-211 for evidence of component leakage or structural distress with the system under pressure as specified in ISC-520(b), when the system is undergoing either a periodic system performance test or a system pressure test. (2) For insulated components, the examinations will be conducted without the removal of the insulation. (3) Components in letdown portion of CVCS system (from pressure control valves to volume control tank, which normally operates at less than 275 psig 200 F) will be visually examined in accordance with ISC-261(c).
C1.6		ISC-261(d) Components	Visual	Cumulative 100% of the required examinations (see Note (c)).	(1) See Note (g).

SECTION C2. PIPING

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C2.1 C2.1.1	C-F	Circumferential butt welds	Volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) The areas of examination include circumferential butt welds at structural discontinuities in piping exceeding 4-inch nominal pipe size and circumferential butt welds in piping exceeding 4-inch nominal pipe size within 3 pipe-diameters of the centerline of rigid pipe anchors, or anchors at the penetration of the primary containment, or at rigidly anchored components. (2) See Note (b). (3) See Note (h).

TABLE 4.4-7 (Cont'd)

SECTION C2. PIPING

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C2.1 C2.1.2	C-G	Circumferential butt welds	Volumetric	One-half of the components in this category will be selected and included in the overall total of required examinations for ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) The areas of examination include circumferential butt welds at structural discontinuities in piping exceeding 4-inch nominal pipe size and circumferential butt welds in piping exceeding 4-inch nominal pipe size within 3 pipe-diameters of the centerline of rigid pipe anchors, or anchors at the penetration of the primary containment, or at rigidly anchored components. (2) A representative sampling among the total number of welds will be selected for examination. (3) See Note (b). (4) See Note (i).
C2.2 C2.2.1	C-F	Longitudinal weld joints in fittings	Volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) components. A cumulative 25% of the overall required examinations will be completed during the interval. See Note (a).	(1) The areas of examination include the longitudinal weld joints in pipe fittings exceeding 4-inch nominal pipe size. (2) See Note (b). (3) See Note (h).

TABLE 4.4-7 (Cont'd)

SECTION C2. PIPING

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C2.2 C2.2.2	C-G	Longitudinal weld Joints in fittings	Volumetric	See Remarks.	Not applicable for Note (i) systems.
C2.3 C2.3.1	C-F	Branch pipe-to-pipe weld joints	Volumetric	See Remarks.	Not applicable for Note (h) systems.
C2.3.2	C-G	Branch pipe-to-pipe weld joints	Volumetric	One-half of the components in this category will be selected and included in the overall total of required examinations for ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) The areas of examination include branch connection weld joints exceeding 4-inch nominal pipe size. (2) A representative sampling among the total number of welds will be selected for examination. (3) See Note (b). (4) See Note (i).

TABLE 4.4-7 (Cont'd)

SECTION C2. PIPING

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C2.4	C-C	Integrally-welded support-to-pipe welds	Suface	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) See Note (b). (2) See Notes (h) and (i).
C2.5	C-D	Pressure-retaining bolting	Visual and either surface or volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) componetns. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) Bolting exceeding 1-inch in diameter will be examined in-place in the bolted-up condition or whenever the bolted connection is disassembled. Flange ligaments between threaded stud holes will be included whenever the connection is disassembled, and feasible examination methods have been developed. (2) A required examination consists of the following: (a) Visual examinations will include 100% of the bolts, studs, and nuts in the bolted joint. (b) Surface examinations will be performed on of the bolting in each joint, but not less than 2 bolts or studs per joint whenever the connection is disassembled. (3) See Note (b). (4) See Notes (h) and (i).

TABLE 4.4-7 (Cont'd)

SECTION C2. PIPING

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C2.6	C-E	Supports and hangers	Visual	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Notes (a) and (e).	(1) See Note (b).
C2.7	---	ISC-261(c) System	Visual	Cumualtive 100% of the required examinations. See Note (c).	(1) Examinations will be conducted in accordance with the procedure of IS-211 for evidence of component leakage or structural distress with the system under pressure as specified in ISC-520(b), when the system is undergoing either a periodic system performance test or a system pressure test. (2) For insulated components, the examination will be conducted without the removal of the insulation.

TABLE 4.4-7 (Cont'd)

SECTION C2. PIPING

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
					(3) Components in letdown portion of CVCS system (from pressure control valves to volume control tank, which normally operates at less than 275 psig 200 F) will be visually examined in accordance with ISC-261(c).
C2.8	---	ISC-261(d) systems	Visual	Cumulative 100% of the required examinations (see Note (c)).	(1) See Not (g).
				SECTION C3. PUMPS	
C3.1 C.3.1.1	C-F	Pump casing welds	Volumetric	See Remarks.	Not applicable for Note (j) pumps.
C.3.1.2	C-G	Pump casing welds	Volumetric	See Remarks.	Not applicable for Note (j) pumps.

TABLE 4.4-7 (Cont'd)

SECTION C3. PUMPS

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C3.2	C-H	Pump Casing	Visual	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) See Note (b). (2) See Note (j).
C3.3	C-D	Pressure-retaining bolting	Visual and either surface or volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) Bolting exceeding 1-inch in diameter will be examined in-place in the bolted-up condition or whenever the bolted connection is disassembled. Flange ligaments between threaded stud holes will be included whenever the connection is disassembled, and feasible examination methods have been developed. (2) A required examination consists of the following: (a) Visual examinations will include 100% of the bolts, studs, and nuts in the bolted joint. (b) Surface examinations will be performed on 10% of the bolting in each joint, but not less than 2 bolts or studs per joint whenever the connection is disassembled. (3) See Note (b). (4) See Note (j).

TABLE 4.4-7 (Cont'd)

SECTION C3. PUMPS

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C3.4	C-E	Supports and hangers	Visual	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Notes (a) and (e).	(1) See Note (b). (2) See Note (j).
C3.5	---	Integrally welded supports	Surface	See Remarks	Not applicable for Note (j) pumps.
C3.6	---	ISC-261(c) components	Visual	Cumulative 50% of the required examinations See Note (c).	(1) Examinations will be conducted in accordance with the procedures of IS-211 for evidence of component leakage or structural distress with the system under pressure as specified in ISC-520(b), when the system is undergoing either a periodic system performance test or a system pressure test. (2) For insulated components, the examinations will be conducted without the removal of the insulation.

TABLE 4.4-7 (Cont'd)

SECTION C3. PUMPS

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval		Remarks
					(3)	Components in letdown portion of CVCS system (from pressure control valves to volume control tank, which normally operates at less than 275 psig 200 F) will be visually examined in accordance with ISC-261(c).
C3.7	---	ISC-261(d) Components	Visual	Cumulative 50% of the required examinations (see Note (c)).	(1)	See Note (g).

SECTION C4. VALVES

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval		Remarks
C4.1 C4.1.1	C-F	Valve body welds	Volumetric	See Remarks		Not applicable for Notes (h) and (i) systems.
C4.1.2	C-G	Valve body welds	Volumetric	See Remarks		Not applicable for Notes (h) and (i) systems.

TABLE 4.4-7 (Cont'd)

SECTION C4. VALVES

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C4.2	C-H	Valve Bodies	Visual	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) See Note (b). (2) See Notes (h) and (i).
C4.3	C-D	Pressure-retaining bolting	Visual and either surface or volumetric	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Note (a).	(1) Bolting exceeding 1-inch in diameter will be examined in-place in the bolted-up condition or whenever the bolted connection is disassembled. Flange ligaments between threaded stud holes will be included whenever the connection is disassembled. See Note (d). (2) A required examination consists of the following: (a) Visual examinations will include 100% of the bolts, studs, and nuts in the bolted joint. (b) Surface examinations will be performed on 10% of the bolting in each joint, but not less than 2 bolts or studs per joint whenever the connection is disassembled. (3) See Note (b). (4) See Notes (h) and (i).

TABLE 4.4-7 (Cont'd)

SECTION C4. VALVES

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C4.4	C-E	Supports and hangers	Visual	The components in this category are included in the overall total of required examinations for ISC-261(a) and ISC-261(b) components. A cumulative 25% of the overall total of required examinations will be completed during the interval. See Notes (a) and (e).	(1) See Note (b).
C4.5	---	Integrally-welded supports	Surface	See Remarks	Not applicable for Notes (h) and (i) systems.
C4.6	---	ISC-261(c) components	Visual	Cumulative 50% of the required examinations. See Note (c).	(1) Examinations will be conducted in accordance with the procedures of IS-211 for evidence of component leakage or structural distress with the system under pressure as specified in ISC-520(b), when the system is undergoing either a periodic system performance test or a system pressure test. (2) For insulated components, the examinations will be conducted without the removal of the insulation. (3) Components in letdown portion of CVCS system (from pressure control valves to volume control tank, which normally operates at less than 275 psig 200 F) will be visually examined in accordance with ISC-261(c).

TABLE 4.4-7 (Cont'd)

SECTION C4. VALVES

Item No.	Category	Examination Area	Examination Method	Tentative Inspection During 10-year Interval	Remarks
C4.7	---	ISC-261(d) Components	Visual	Cumulative 50% of the required examinations (see note (c)).	(1) See Note (g).

Note (a): At least part of the overall total of required examinations will be performed by the expiration of 1/3, 2/3, and the end of the 10-year interval.

(b): The total number of required examinations is dependent upon whether or not a system consists of multiple streams which perform the same (or redundant) functions as stated in ISC-242(a); the distribution of individual components to be examined will be determined in accordance with ISC-242(b), (c), and (d).

(c): At least 25% of the required examinations will be completed by the expiration of 1/3 of the 10-year inspection interval, with not more than 66-2/3% of the required examinations completed by the expiration of 2/3 of the 10-year inspection interval.

(d): Present ultrasonic techniques and procedures may not be amenable to examination of pump castings.

(e): Where accessibility to welds prohibits their examination, each weld will be evaluated on a case by case basis.

(f): The regenerative, letdown, and shutdown heat exchangers, and the steam generators (secondary side) are to be examined.

(g): Components and systems in which the contained fluid will be verified by periodic sampling and tests.

(h): Portions of the shutdown cooling systems are to be examined.

(i): Portions of the main steam and feedwater systems are to be examined.

(j): Low pressure safety injection pumps are to be examined.

SAFETY CLASS 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.3 The structural integrity of components identified in Section 3.2.2 of the FSAR as Safety Class 3 components shall be maintained at a level consistent with the acceptance criteria in Specification 4.4.10.3.

APPLICABILITY: ALL MODES.

ACTION:

With the structural integrity of any of the above components not conforming to the above requirements, restore the structural integrity of the component to within its limit or isolate the affected components from service. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.3 The following inspection program shall be performed:

a. Inservice Inspections The structural integrity of the Safety Class 3 components shall be demonstrated at least once per 40 months during periods of normal reactor operation or during system performance testing by verifying via visual inspections, as outlined by the inspection program shown in Table 4.4-8, that there is no evidence of unanticipated component leakage, structural distress, or corrosion.

An initial report of any abnormal degradation of the structural integrity of the Safety Class 3 components detected during the above required inspections shall be made within 10 days after detection and the detailed report shall be submitted pursuant to Specification 6.9.1 within 90 days after completion of the surveillance requirements of this specification.

SURVEILLANCE REQUIREMENTS (Continued)

b. System Pressure Tests The structural integrity of the Safety Class 3 components shall be demonstrated at least once per 10 years by performing system pressure tests at the following test pressures:

1. For closed systems, at least 110 percent of the design pressure,

2. For open storage tanks, at least the nominal hydrostatic pressure developed with the tanks filled to design capacity, and

3. Open-ended portions of systems may be exempted from pressure testing.

c. Pipe Hanger Inspections The structural integrity of the Safety Class 3 components shall be demonstrated at least once per 40 months by verifying via visual inspections that the supports and hangers for piping and components over 4 inches in diameter show no evidence of inadequate support, unintended restraint, or structural distress.

TABLE 4.4-8

INSERVICE INSPECTION PROGRAM - SAFETY CLASS 3 COMPONENTS

1. Class 3 piping greater than 4 inches and components with pipe connections greater than 4 inches will be pressure tested and visually examined near or at the end of the 10-year inspection interval. In addition, components will be visually examined during periods of normal operations or during system performance testing once during each 1/3 of the year 10-year inspection interval.

2. Open-ended portions of systems will not be pressure tested.

CORE BARREL MOVEMENT

LIMITING CONDITION FOR OPERATION

3.4.11 Core barrel movement shall be limited to less than the Amplitude Probability Distribution (APD) and Spectral Analysis (SA) Alert Levels for the applicable THERMAL POWER level.

APPLICABILITY: MODE 1.

ACTION:

a. With the APD and/or SA exceeding their applicable Alert Levels, POWER OPERATION may proceed provided the following actions are taken:

   1. APD shall be measured and processed at least once per 24 hours,

   2. SA shall be measured within 48 hours and at least once per 24 hours thereafter and SA shall be processed at least once per 7 days, and

   3. A Special Report, identifying the cause(s) for exceeding the applicable Alert Level, shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days of detection.

b. With the APD and/or SA exceeding their applicable Action Levels, within 24 hours reduce THERMAL POWER BY $\geq$ 25% of RATED THERMAL POWER and demonstrate, through monitoring of the excore neutron detectors, that APD and SA have been reduced to below their applicable Alert Level limits or be in HOT STANDBY within the next 6 hours.

c. With the measured levels of APD and/or SA differing from their baseline levels by more than 10%, a Special Report describing the measured levels shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 10 days of data processing.

CORE BARREL MOVEMENT (Continued)

SURVEILLANCE REQUIREMENTS

4.4.11.1 Baseline Monitoring Core barrel movement Alert Levels and Action Levels, as determined by APD and SA monitoring of the excore neutron detectors, shall be determined at nominal THERMAL POWER levels of 20%, 50%, 80% and 100% of RATED THERMAL POWER during the reactor startup test program; these Alert Levels and Action Levels shall be reported in a Special Report pursuant to Specification 6.9.2 within 31 days after initially reaching 100% of RATED THERMAL POWER.

4.4.11.2 Routine Monitoring Core barrel movement shall be determined to be less than the APD and SA Alert Levels by using the excore neutron detectors to measure APD and SA at the following frequencies:

a. APD data shall be measured and processed at least once per 7 days.

b. SA data shall be measured and processed at least once at nominal THERMAL POWER levels of 20%, 50%, 80% and 100% of RATED THERMAL POWER after each refueling and at least once per 4 months thereafter.

4.4.11.3 Reports The results of all periodic APD and SA monitoring shall be included in the Annual Operating Report for the period in which the monitoring was performed.

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. Between 1090 and 1170 cubic feet of borated water,

c. A minimum boron concentration of 1720 PPM, and

d. A nitrogen cover-pressure of between 200 and 250 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one safety injection tank inoperable, except as a result of a closed isolation valve, restore the inoperable tank to OPERABLE status within one hour or be in HOT SHUTDOWN within the next 8 hours.

b. With one safety injection tank inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 8 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each safety injection tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the water level and nitrogen cover-pressure in the tanks, and

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

* With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of $\geq$ 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 1750 psia, by verifying that power to the isolation valve operator is removed by maintaining the breaker open under administrative control.

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

    1. When the RCS pressure exceeds 350 psia, and

    2. Upon receipt of a safety injection test signal.

ECCS SUBSYSTEMS - $T_{avg} \geq 300°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE high-pressure safety injection (HPSI) pump (one ECCS subsystem shall include HPSI pump A and the second ECCS subsystem shall include either HPSI pump B or C),

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.

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 HOT SHUTDOWN within the next 12 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.

*With pressurizer pressure $\geq$ 1750 psia.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

Valve Number	Valve Function	Valve Position
1. V-3659	1. Mini-flow isolation	1. Open
2. V-3660	2. Mini-flow isolation	2. Open

b. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each high-pressure safety injection pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure of $\geq$ 1138 psig on recirculation flow.

   c) Operates for at least 15 minutes.

2. Verifying that each low-pressure safety injection pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure of $\geq$ 175 psig on recirculation flow.

   c) Operates for at least 15 minutes.

3. Verifying that upon a recirculation actuation signal, the containment sump isolation valves open.

4. Cycling each testable, power operated valve in the flow path through at least one complete cycle of the full travel.

SURVEILLANCE REQUIREMENTS (Continued)

5. 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.

6. Verifying that each ECCS subsystem is aligned to receive electrical power from separate OPERABLE emergency busses.

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

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

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

d. At least once per 18 months by:

1. Verifying automatic isolation of the shutdown cooling system from the Reactor Coolant System when the Reactor Coolant System pressure is above 300 psig.

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 86 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 0.5 ± 0.1 lbs of TSP from a TSP storage basket is submerged, without agitation, in 50 ± 1.0 gallons of $200 \pm 10°F$ borated water from the RWT, the pH of the mixed solution is raised to ≥ 6 within 4 hours.

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation Signal.

3. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Signal;

    a. High-Pressure Safety Injection pump.

    b. Low-Pressure Safety Injection pump.

ECCS SUBSYSTEMS - T_{avg} < 300°F

LIMITING CONDITION FOR OPERATION

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

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

b. An OPERABLE flow path capable of taking suction from the refueling water tank on a safety injection actuation signal and automatically transferring suction to the containment sump on a recirculation actuation signal.

APPLICABILITY: MODES 3* and 4.

ACTION:

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

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

SURVEILLANCE REQUIREMENTS

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

*With pressurizer pressure < 1750 psia.

REFUELING WATER TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The refueling water tank shall be OPERABLE with:

a. A minimum contained volume 371,800 gallons of borated water,

b. A minimum boron concentration of 1720 ppm,

c. A maximum water temperature of 100°F,

d. A minimum water temperature of 55°F when in MODES 1 and 2, and

e. A minimum water temperature of 40°F when in MODES 3 and 4.

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 water level in the tank, and

2. Verifying the boron concentration of the water.

b. At least once per 24 hours by verifying the RWT temperature.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 CONTAINMENT VESSEL

CONTAINMENT VESSEL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 CONTAINMENT VESSEL INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that:

1. All containment vessel 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 3.6.3.1, and

2. All containment vessel equipment hatches are closed and sealed.

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

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. $\leq L_a$, 0.50 percent by weight of the containment air per 24 hours at P_a, (39.6 psig), or

2. $\leq L_t$, 0.32 percent by weight of the containment air per 24 hours at a reduced pressure of P_t, (19.8 psig).

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests as identified in Table 3.6-1 when pressurized to P_a.

c. A combined leakage rate of $\leq 0.12\ L_a$ 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 leakage rate(s) to within the limit(s) 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 provisons of ANSI N45.4-1972:

a. Three Type A tests (Overall Integrated Containment Leakage Rate) shall be conducted at 40 $\pm$ 10 month intervals during shutdown at either P_a (39.6 psig) or at P_t (19.8 psig) during each 10-year

SURVEILLANCE REQUIREMENTS (Continued)

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 between the Type A test and the supplemental test.

   3. Requires the quantity of gas injected into the containment or bled from the containment during the supplemental test to be equivalent to at least 25 percent of the total measured leakage rate at P_a (39.6 psig) or P_t (19.8 psig).

d. Type B and C tests shall be conducted with gas at P_a (39.6 psig) at intervals no greater than 24 months except for tests involving air locks.

e. The combined bypass leakage rate shall be determined to be < 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 (39.6 psig) during each Type A test.

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

SURVEILLANCE REQUIREMENTS (Continued)

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

TABLE 3.6-1

CONTAINMENT LEAKAGE PATHS

Penetration	System	Valve Tag Number	Location to Containment	Service	Test Type*
7	Makeup Water	Gate (I-MV-15-1) Check (I-V-15-1347)	Outside Inside	Primary Makeup Water	Bypass
8	Station Air	Globe (I-V-18-947) Globe (I-V-18-947)	Outside Outside	Station Air Supply	Bypass
9	Instrument Air	Gate (I-MV-18-1) Check (I-V-18-957)	Outside Inside	Instrument Air Supply	Bypass
10	Containment Purge	Butterfly (I-FCV-25-4) Butterfly (I-FCV-25-5)	Inside Outside	Containment Purge Exhaust	Type C
11	Containment Purge	Butterfly (I-FCV-25-3) Butterfly (I-FCV-25-2)	Inside Outside	Containment Purge Supply	Type C
14	Waste Management	Globe (V-6741) Check (V-6779)	Outside Outside	Nitrogen supply to SI Tanks	Bypass
23	Component Cooling	Butterfly (I-HCV-14-7) Butterfly (I-HCV-14-1)	Outside Outside	RC Pump CW supply	Bypass
24	Component Cooling	Butterfly (I-HCV-14-6) Butterfly (I-HCV-14-2)	Outside Outside	RC Pump CW Return	Bypass
25	Fuel Transfer Tube	Blind Flange	Inside	Fuel Transfer	Bypass
26	CVCS	Globe (V-2515) Globe (V-2516)	Inside Inside	Letdown Line	Bypass
28	Sampling	Globe (V-5200) Globe (V-5203)	Outside Outside	Reactor Coolant Sample	Bypass

TABLE 3.6-1 (Continued)

Penetration	System	Valve Tag Number	Location to Containment	Service	Test Type*
29	Sampling	Globe (V-5202) Globe (V-5205)	Outside Outside	Pressurizer Steam Space Sample	Bypass
29	Sampling	Globe (V-5201) Globe (V-5204)	Outside Outside	Pressurizer Surge Line Sample	Bypass
31	Waste Management	Gate (V-6554) Gate (V-6555)	Outside Outside	Containment Vent Header	Bypass
41	Safety Injection Tank Test Lines	Gate (V-3463) Gate (I-V-03-1307)	Outside Outside	Safety Injection Tank Fill and Sampling	Bypass
42	Waste Management	Gate (I-LCV-07-11A) Gate (I-LCV-07-11B)	Outside Outside	Reactor Cavity Sump Pump Discharge	Bypass
43	Waste Management	Gate (V-6301) Gate (V-6302)	Outside Outside	Reactor Drain Tank Pump Suction	Bypass
44	CVCS	Gate (V-2505) Gate (I-SE-01-1)	Outside Inside	RC Pump Controlled Bleedoff	Bypass
46	Fuel Pool Cleanup	Gate (I-V-07-206) Gate (I-V-07-189)	Outside Inside	Refueling Cavity Purification Flow Inlet	Bypass
47	Fuel Pool Cleanup	Gate (I-V-07-170) Gate (I-V-07-188)	Outside Inside	Refueling Cavity Purification Flow Outlet	Bypass
48	Sampling	Globe (I-FSE-27-01,02, 03,04) Globe (I-FSE-27-08)	Inside Outside	H_2 Sampling	Type C

TABLE 3.6-1 (Continued)

Penetration	System	Valve Tag Number	Location to Containment	Service	Test Type*
48	Sampling	Globe (I-FSE-27-11) Check (I-FSE-27-1341)	Outside Inside	H_2 Sampling	Type C
51	Sampling	Globe (I-FSE-27-5,6,7) Globe (I-FSE-27-9)	Inside Outside	H_2 Sampling	Type C
51	Sampling	Globe (I-FSE-27-10) Check (I-FES-27-1342)	Outside Inside	H_2 Sampling	Type C
52a	Sampling	Gate (I-FCV-26-1) Gate (I-FCV-26-2)	Inside Outside	Radiation Monitoring	Bypass
52b	Sampling	Gate (I-FCV-26-3) Gate (I-FCV-26-4)	Inside Outside	Radiation Monitoring	Bypass
52c	Sampling	Gate (I-FCV-26-5) Gate (I-FCV-26-6)	Inside Outside	Radiation Monitoring Return	Bypass
52d	ILRT	Gate (I-V00140(1325)) Gate (I-V00143(1325))	Inside Outside	ILRT Test Tap	Bypass
52e	ILRT	Gate (I-V00139(1322)) Gate (I-V00144(1322))	Inside Outside	ILRT Test Tap	Bypass
54	ILRT	Blind Flange Gate (I-V00101(612))	Inside Outside	ILRT Pressure Connection	Bypass
56	Containment H_2 Purge	Gate (V-25-11) Gate (V-25-12)	Outside Outside	Hydrogen Purge Outside Air Makeup	Bypass
57	Containment H_2 Purge	Gate (V-25-13) Gate (V-25-14)	Outside Outside	Hydrogen Purge Exhaust	Bypass

TABLE 3.6-1 (Continued)

Penetration	System	Valve Tag Number	Location to Containment	Service	Test Type*
58	Containment H_2 Purge	Gate (V-25-15) Gate (V-25-16)	Outside Outside	Hydrogen Purge Exhaust	Bypass
67	Vacuum Relief	Check (I-V-25-20) Butterfly (I-FCV-25-7)	Inside Outside	Containment Vacuum Relief	Type C
68	Vacuum Relief	Check (I-V-25-21) Butterfly (I-FCV-25-8)	Inside Outside	Containment Vacuum Relief	Type C
Personnel Lock	N.A.	None	N.A.	Ingress & Egress to Containment	Type B**
Escape Lock	N.A.	None	N.A.	Emergency Ingress & Egress to Containment	Type B**
Maintenance Hatch	N.A.	None	N.A.	Vessel Maintenance	Type B (Gasket Interspace)
Electrical Penetrations	N.A.	All primary canisters except welded spares	N.A.	Electrical connections in PCV	Type B
1	Main Steam Steel Containment Nozzles	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

TABLE 3.6-1 (Continued)

Penetration	System	Valve Tag Number	Location to Containment	Service	Test Type*
4	Feedwater Steel Containment Nozzles	Tap 1 Tap 2	Outside Outside	Expansion Bellows	Type B
25	Fuel Tube Steel Containment Nozzles	Tap 1 Tap 2	Inside Outside	Expansion Bellows	Type B

* Type C and bypass tests are conducted in the same manner, the only difference is in the acceptance criteria that is applicable.
** In accordance with Specification 4.6.1.3.b.

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 $\leq 0.05\ L_a$ at P_a, (39.6 psig).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With an air lock inoperable, except as a result of an inoperable door gasket, restore the 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.

b. With an air lock inoperable due to an inoperable door gasket:

1. Maintain the remaining door of the affected air lock closed and sealed, and

2. Restore the air lock to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a. After each opening by verifying the seal leakage is $< 0.01\ L_a$ as determined by precision flow measurement when measured for at least 30 seconds with:

1. The volume between the personnel air lock seals at a constant pressure of 39.6 psig, and

2. The volume between the emergency air lock seals at a constant pressure of 10 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 6 months by conducting an overall air lock leakage test at P_a (39.6 psig) and by verifying that the overall air lock leakage rate is within its limit, and

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

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -0.7 and 2.4 PSIG.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

CONTAINMENT SYSTEMS

AIR TEMPERATURE

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 > 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 three of the following locations and shall be determined at least once per 24 hours:

Location

a. Containment fan cooler No. 1A air intake, elevation 45 feet.

b. Containment fan cooler No. 1B air intake, elevation 45 feet.

c. Containment fan cooler No. 1C air intake, elevation 62 feet.

d. Containment fan cooler No. 1D air intake, elevation 45 feet.

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 Specification 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 Specification 4.6.1.2) by a visual inspection of the accessible interior and exterior surfaces of the vessel and verifying no apparent changes in appearance of the surfaces or other abnormal degradation. An initial report of any abnormal degradation of the containment vessel detected during the above required inspections shall be made within 10 days after detection and the detailed report shall be submitted to the Commission pursuant to Specification 6.9.1 within 90 days after completion of the surveillance requirements of this specification.

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 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:

a. With one containment spray system inoperable and all four containment fan coolers OPERABLE, restore the inoperable spray system to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

b. With one containment spray system inoperable and one containment fan cooler inoperable, restore either the inoperable spray system or the inoperable fan cooler to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Starting each spray pump from the control room,

2. Verifying, that on recirculation flow, each spray pump develops a discharge pressure of $\geq$ 200 psig,

3. Verifying that each spray pump operates for at least 15 minutes,

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

SURVEILLANCE REQUIREMENTS (Continued)

4. Cycling each testable, power operated or automatic valve in the flow path through at least one complete cycle of full travel,

5. Verifying that upon a recirculation actuation signal, the containment sump isolation valves open and that a recirculation mode flow path via an OPERABLE shutdown cooling heat exchanger is established, and

6. Verifying that each valve (manual, power operated or automatic) in the flow path is positioned to take suction from the RWT on a Containment Pressure--High-High signal.

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

1. Cycling each power operated (excluding automatic) valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve in the flow path actuates to its correct position on a Containment Pressure--High-High signal.

3. Verifying that each spray pump starts automatically on a Containment Pressure--High-High signal.

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

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Four containment fan coolers shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one containment fan cooler inoperable and both containment spray systems OPERABLE, restore the inoperable fan cooler to OPERABLE status within 30 days or be in HOT SHUTDOWN within the next 12 hours.

b. With one containment fan cooler inoperable and one containment spray system inoperable, restore either the inoperable fan cooler or the inoperable spray system to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 Each containment fan cooler shall be demonstrated OPERABLE at least once per 31 days on a STAGGERED TEST BASIS by:

a. Starting each unit from the control room,

b. Verifying that each unit operates for at least 15 minutes, and

c. Verifying a cooling water flow rate of $\geq$ 1200 gpm to each cooling unit.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-2 inoperable, 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

4.6.3.1.1 The isolation valves specified in Table 3.6-2 shall be demonstrated OPERABLE:

a. At least once per 92 days by cycling each power operated or automatic valve testable during plant operation through at least one complete cycle of full travel.

SURVEILLANCE REQUIREMENTS (Continued)

b. Immediately 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 the cycling test, above, and verification of isolation time.

4.6.3.1.2 Each 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, each isolation valve actuates to its isolation position,

b. Cycling each power operated or automatic valve through at least one complete cycle of full travel and measuring its isolation time, and

c. Cycling each manual valve not locked, sealed or otherwise secured in the closed position through at least one complete cycle of full travel.

TABLE 3.6-2

CONTAINMENT ISOLATION VALVES

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Isolation Time (Sec)
A. CONTAINMENT ISOLATION				
1. I-FCV-25-4,5	10	Containment purge air exhaust, CIS	No	5
2. I-FCV-25-2,3	11	Containment purge supply, CIS	No	5
3. I-MV-15-1	7	Primary makeup water, CIS	Yes	19
4. I-MV-18-1	9	Instrument air supply, CIS	No	28
5. V-6741	14	Nitrogen supply to safety injection tanks, CIS	Yes	5
6. I-HCV-14-1 & 7	23	Reactor coolant pump cooling water supply, SIAS	No	5
7. I-HCV-14-6 & 2	24	Reactor coolant pump cooling water return, SIAS	No	5
8. V-2515,2516	26	Letdown line, CIS, SIAS	No	5
9. V-5200,5203	28	Reactor coolant sample, CIS	Yes	5
10. V-5201,5204	29	Pressurizer surge line sample, CIS	Yes	5
11. V-5202,5205	29	Pressurizer steam space sample, CIS	Yes	5
12. V-6554,6555	31	Containment vent header, CIS	Yes	5
13. I-LCV-07-11A,11B	42	Reactor cavity sump pump discharge, CIS	Yes	10
14. V-6301,6302	43	Reactor drain tank pump suction, CIS	Yes	5
15. V-2505	44	Reactor coolant pump controlled bleedoff, CIS	No	5
16. I-SE-01-1	44	Reactor coolant pump controlled bleedoff, CIS	No	5

TABLE 3.6-2 (Continued)

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Isolation Time (Sec)
B. MANUAL OR REMOTE MANUAL				
1. I-V-18-947	8	Station air supply, Manual	Yes	NA
2. I-V-25-11,12	56	Hydrogen purge outside air make-up, Manual (NC)	Yes	NA
3. I-V-25-13,14, 15,16	57 & 58	Hydrogen purge exhaust, Manual (NC)	Yes	NA
4. V-3463	41	Safety injection tank test line, Manual (NC)	Yes	NA*
5. I-V-03-1307	41	Safety injection tank test line, Manual (NC)	Yes	NA*
6. V-07206, V-07189	46	Refueling cavity purification flow inlet, Manual (NC)	Yes	NA
7. V-07170, V-07188	47	Refueling cavity purification flow outlet, Manual (NC)	Yes	NA
8. I-FSE-27-1,2,3, 4,8,10	48	Hydrogen sampling line, Remote manual	Yes	NA*
9. I-FSE-27-5,6,7, 9,11	51	Hydrogen sampling line, Remote manual	Yes	NA*

TABLE 3.6-2 (Continued)

Valve Tag Number	Penetration Number	Function	Testable During Plant Operation	Isolation Time (Sec)
10. I-FCV-26-1 & 2	52a	Radiation monitoring	Yes	NA
11. I-FCV-26-3 & 4	52b	Radiation monitoring	Yes	NA
12. I-FCV-26-5 & 6	52c	Radiation monitoring, return	Yes	NA
13. I-V00140(1325) I-V00143(1325)	52d	ILRT test tap	Yes	NA
14. I-V00139(1322) I-V00144(1322)	52e	ILRT test tap	Yes	NA
15. I-V00101(612)	54	ILRT pressure connection	Yes	NA

NA - Manual Valve-Isolation time not applicable.
* May be opened on an intermittent basis under administrative control.
** Normally closed valves - Isolation time not applicable.

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 The containment hydrogen analyzer and grab sample system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With either the hydrogen analyzer or grab sample system inoperable, restore the inoperable analyzer or grab sample system to OPERABLE status within 30 days or be in at least HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.6.4.1.1 The hydrogen analyzer shall be demonstrated OPERABLE at least once per 92 days by:

a. Performing a CHANNEL CALIBRATION using sample gases containing:

   1. One volume percent hydrogen, balance nitrogen, and

   2. Four volume percent hydrogen, balance nitrogen.

b. Verifying that the analyzer is aligned to receive electrical power from an OPERABLE emergency bus.

4.6.4.1.2 The grab sample system shall be demonstrated OPERABLE at least once per 92 days by using the hydrogen sample pumps to draw a sample of the containment atmosphere into the grab sample canister. The hydrogen sample pumps shall be used on an alternating basis.

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 $\geq$ 700°F within 90 minutes and is maintained for at least 2 hours.

b. At least once per 18 months by:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying during a recombiner system functional test that the heater sheath temperature increases to $\geq$ 1200°F within 5 hours and is maintained for at least 4 hours.

4. Verifying the integrity of the heater electrical circuits by performing a continuity and resistance to ground test immediately following the above required functional test. The resistance to ground for any heater phase shall be $\geq$ 10,000 ohms.

3/4.6.5 VACUUM RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.6.5.1 The containment vessel to annulus vacuum relief valves shall be OPERABLE with an actuation setpoint of 2.25 ± 0.25 inches Water Gauge differential.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one 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

4.6.5.1 The containment vessel to annulus vacuum relief valves shall be demonstrated OPERABLE:

a. At least once per 6 months by verifying valve partial opening (≥ 5 percent of valve full travel) and that valve operation is not restricted by corrosion, dirt, wear or debris.

b. At least once per 3 years by verifying that the valves open fully within 8 seconds at 2.25 ± 0.25 inches Water Gauge differential.

3/4.6.6 SECONDARY CONTAINMENT

SHIELD BUILDING VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.6.1. Two independent shield building ventilation systems shall be OPERABLE.

APPLICABLILITY: MODES 1, 2, 3 and 4.

ACTION:

With one shield building 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

4.6.6.1 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 filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

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

   1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

   2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested accordance with ANSI N510-1975 (130°C, 95% R.G.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of system operation by either:

1. Verifying that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.); or

2. Verifying that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510-1975 (130°C, 95% R.H.) and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

SURVEILLANCE REQUIREMENTS (Continued)

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%, and

b) Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

d. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is $\leq$ 6.15 inches Water Gauge while operating the ventilation system at a flow rate of 6000 cfm $\pm$ 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that the filtration system starts automatically on a Containment Isolation Signal (CIS).

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

5. Verifying that each system produces a negative pressure of $\geq$ 2.0 inches W.G. in the annulus within 2 minutes after a Containment Isolation Signal (CIS).

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the filtration system at a flow rate of 6000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the filtration system at a flow rate of 6000 cfm $\pm$ 10%.

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.

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 Specification 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 prior to increasing the Reactor Coolant System temperature above 200°F.

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 Specification 4.6.1.2) by a visual inspection of the accessible interior and exterior surfaces of the shield building and verifying no apparent changes in appearance of the concrete surfaces or other abnormal degradation. An initial report of any abnormal degradation of the shield building detected during the above required inspections shall be made within 10 days after detection and the detailed report shall be submitted to the Commission pursuant to Specification 6.9.1 within 90 days after completion of the surveillance requirements of this specification.

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.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.1.1 Each main steam line code safety valve shall be demonstrated OPERABLE, with lift settings and orifice sizes as shown in Table 4.7-1, in accordance with Section XI of the ASME Boiler and Pressure Vessel Code, 1974 Edition.

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL-HIGH TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES DURING OPERATION WITH BOTH STEAM GENERATORS

Maximum Number of Inoperable Safety Valves on Any Operating Steam Generator	Maximum Allowable Power Level-High Trip Setpoint (Percent of RATED THERMAL POWER)
1	93.2
2	79.8
3	66.5

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	Header A	Header B	LIFT SETTING (± 1%)	ORIFICE SIZE
a.	8201	8205	1000 psia	16 $in.^2$
b.	8202	8206	1000 psia	16 $in.^2$
c.	8203	8207	1000 psia	16 $in.^2$
d.	8204	8208	1000 psia	16 $in.^2$
e.	8209	8213	1040 psia	16 $in.^2$
f.	8210	8214	1040 psia	16 $in.^2$
g.	8211	8215	1040 psia	16 $in.^2$
h.	8212	8216	1040 psia	16 $in.^2$

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 motor driven feedwater pumps, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two motor driven pumps and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each auxiliary feedwater pump shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Starting each pump from the control room,

2. Verifying that:

   a) Each motor driven pump develops a discharge pressure of $\geq$ 1342 psig on recirculation flow, and

   b) The steam turbine driven pump develops a discharge pressure of $\geq$ 1342 psig on recirculation flow.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each pump operates for at least 15 minutes.

4. Cycling each testable power operated or automatic valve in the flow path through at least one complete cycle of full travel.

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

b. At least once per 18 months during shutdown by cycling each power operated valve in the flow path that is not testable during plant operation, through at least one complete cycle of full travel.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank shall be OPERABLE with a minimum contained volume of 116,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With the condensate storage tank inoperable, restore the condensate storage tank 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

4.7.1.3 The condensate storage tank shall be demonstrated OPERABLE at least once per 12 hours by verifying the water level.

PLANT SYSTEMS

ACTIVITY

LIMITING CONDITION FOR OPERATION

3.7.1.4 The specific activity of the secondary coolant system shall be $\leq$ 0.10 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the specific activity of the secondary coolant system > 0.10 μCi/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 4.7-2.

TABLE 4.7-2

SECONDARY COOLANT SYSTEM SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	MINIMUM 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 idicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross activity determination indicates iodine concentrations below 10% of the allowable limit.

MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.5 Each main steam line isolation valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

MODE 1 - With one main steam line isolation valve inoperable, POWER OPERATION may continue provided the inoperable valve is either restored to OPERABLE status or closed within 4 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours.

MODES 2 and 3 - With one main steam line isolation valve inoperable, subsequent operation in MODES 1, 2 or 3 may proceed after the inoperable valve is restored to OPERABLE status or the isolation valve is maintained closed; otherwise, be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

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

b. Verifying full closure within 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 6 seconds need not be determined more often than once per 92 days.

SECONDARY WATER CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.7.1.6 The secondary water chemistry shall be maintained within the limits of Table 3.7-3 by use of All Volatile Treatment (AVT).

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

(To be determined in a manner set forth in the bases in approximately six months and to be imposed by a change to this Specification.)

SURVEILLANCE REQUIREMENTS

4.7.1.6 The secondary water chemistry shall be determined to be within the limits by analysis of those parameters at the frequencies specified in Table 4.7-3.

TABLE 3.7-3

SECONDARY WATER CHEMISTRY LIMITS

Water Sample Location	Parameters*
*	*

* Sample locations, parameters and limits to be established in approximately 6 months based upon test program described in bases.

TABLE 4.7-3

SECONDARY WATER CHEMISTRY SURVEILLANCE REQUIREMENTS

Water Sample Location	Parameters*
*	*

* Sample locations, parameters and frequencies to be established in approximately 6 months based upon test program described in bases.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: ALL MODES.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure of the applicable side to ≤ 200 psig within 30 minutes, and

b. Perform an analysis 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

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

3/4.7.3 COMPONENT COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 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

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

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1. Starting (unless already operating) each pump from the control room.

   2. Verifying that each pump develops at least 93% of the discharge pressure for the applicable flow rate as determined from the manufacturer's Pump Performance Curve.

   3. Verifying that each pump operates for at least 15 minutes.

   4. Verifying that each loop is aligned to receive electrical power from separate OPERABLE emergency busses.

   5. Cycling each testable power operated or automatic valve servicing safety related equipment through at least one complete cycle of full travel.

   6. 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

1. Cycling each power operated (excluding automatic) valve servicing safety related equipment that is not testable during plant operation, through at least one complete cycle of full travel.

2. Verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation Signal.

3/4.7.4 INTAKE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 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

4.7.4.1 At least two intake cooling water loops shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

   1. Starting (unless already operating) each pump from the control room.

   2. Verifying that each pump develops at least 93% of the discharge pressure for the applicable flow rate as determined from the manufacturer's Pump Performance Curve.

   3. Verifying that each pump operates for at least 15 minutes.

   4. Verifying that each loop is aligned to receive electrical power from separate OPERABLE emergency busses.

   5. Cycling each testable power operated or automatic valve servicing safety related equipment through at least one complete cycle of full travel.

   6. 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.

SURVEILLANCE REQUIREMENTS (Continued)

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

   1. Cycling each power operated (excluding automatic) valve servicing safety related equipment that is not testable during plant operation, through at least one complete cycle of full travel.

   2. Verifying that each automatic valve servicing safety related equipment actuates to its correct position on a Safety Injection Actuation Signal.

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. An average water temperature of $\leq$ 96°F.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied, be in at least HOT STANDBY within 1 hour and remove flow barriers and provide cooling water from Big Mud Creek within the next 12 hours.

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 temperature and water level to be within their limits.

4.7.5.1.2 The onsite equipment capability for removing the flow barrier between the intake structure and Big Mud Creek shall be verified at least once per 7 days.

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for the facility site.

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 Specification 4.7.6.1.1 below.

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 10 days following the completion of the survey. The Special Report shall include an evaluation of the facility flood protection if, as evidenced by this survey 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.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 The control room emergency ventilation system shall be OPERABLE with:

a. Two booster fans,

b. Two isolation valves in each outside air intake duct,

c. Two isolation valves in the toilet area air exhaust duct,

d. One filter train, and

e. At least two air conditioning units.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

c. With the filter train inoperable, restore the filter train 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.

d. With only one air conditioning unit OPERABLE, restore at least two air conditioning units to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.7.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is ≤ 120°F.

b. At least once per 31 days by:

1. Initiating flow through the HEPA filter and charcoal adsorber train and verifying that each booster fan operates for at least 15 minutes.

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 housing, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

SURVEILLANCE REQUIREMENTS (Continued)

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 2000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of system operation by either:

1. Verifying that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.); or

2. Verifying that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of ≥ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510-1975 (130°C, 95% R.H.) and the samples are prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%, and

b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4.15 inches Water Gauge while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

2. Verifying that on a containment isolation signal or chlorine accident detection signal, the system automatically isolates the control room within 35 seconds and 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 ≥ 1/8 inch W.G. relative to the outside atmosphere during system operation with ≤ 100 cfm outside air intake.

f. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

g. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

3/4.7.8 ECCS AREA VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 Two independent ECCS area exhaust air filter trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one ECCS area exhaust air filter train inoperable, restore the inoperable train 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

4.7.8.1 Each ECCS area exhaust air filter train 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 filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

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

   1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

   2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

    a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

    b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 30,000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of system operation by either:

1. Verifying that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.); or

2. Verifying that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the samples are tested in accordance with ANSI N510-1975 (130°C, 95% R.H.) and the samples are prepared by either:

    a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

    b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

SURVEILLANCE REQUIREMENTS (Continued)

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 30,000 cfm $\pm$ 10%, and

b) Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

d. At least once per 18 months:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4.15 inches Water Gauge while operating the ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

3. Verifying that the filter train starts on a Safety Injection Actuation Signal.

e. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

f. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 30,000 cfm $\pm$ 10%.

3/4.7.9 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.9.1 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 $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

a. Each sealed source with removable contamination in excess of the above limit shall be immediately withdrawn from use and:

    1. Either decontaminated and repaired, or

    2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.9.1.1 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.

4.7.9.1.2 Test Frequencies - Each category of sealed sources shall be tested at the frequencies described below.

a. Sources in use (excluding startup sources previously subjected to core flux) - At least once per six months for all sealed sources containing radioactive material:

SURVEILLANCE REQUIREMENTS (Continued)

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

2. In any form other than gas.

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

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

4.7.9.1.3 Reports - A Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days if source leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3/4.7.10 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.10.1 All hydraulic snubbers listed in Table 3.7-2 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic snubbers inoperable, restore the inoperable snubber(s) 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

4.7.10.1.1 Each hydraulic snubber with seal material fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment and approved as such by the NRC, shall be determined OPERABLE at least once after not less than 4 months but within 6 months of initial criticality and in accordance with the inspection schedule of Table 4.7-3 thereafter, by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors. Initiation of the Table 4.7-3 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

4.7.10.1.2 Each hydraulic snubber with seal material not fabricated from ethylene propylene or other materials demonstrated compatible with the operating environment shall be determined OPERABLE at least once per 31 days by a visual inspection of the snubber. Visual inspections of the snubbers shall include, but are not necessarily limited to, inspection of the hydraulic fluid reservoirs, fluid connections, and linkage connections to the piping and anchors.

HYDRAULIC SNUBBERS (Continued)

SURVEILLANCE REQUIREMENTS (Continued)

4.7.10.1.3 During shutdown, 18 months after initial criticality and at least once per 18 months thereafter, a representative sample of at least 10 snubbers or at least 10% of all snubbers listed in Table 3.7-2, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers selected for functional testing shall be selected on a rotating basis except snubbers identified in Table 3.7-2 as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
SS-1 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-2 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-3 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-4 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-5 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-6 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-7 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-8 1A	MS, Steam Gen. 1A, Elev. 62'	I	No	Yes
SS-1 1B	MS, Steam Gen. 1B, Elev. 62'	I	No	Yes
SS-2 1B	MS, Steam Gen. 1B, Elev. 62'	I	No	Yes
SS-3 1B	MS, Steam Gen. 1B, Elev. 62'	I	No	Yes
SS-4 1B	MS, Steam Gen. 1B, Elev. 62'	I	No	Yes
SS-5 1B	MS, Steam Gen. 1B, Elev. 62'	I	No	Yes
SS-6 1B	MS, Steam Gen. 1B, Elev. 62'	I	No	Yes
SS-7 1B	MS, Steam Gen. 1B, Elev. 62'	I	No	Yes
SS-8 1B	MS, Steam Gen. 1B, Elev. 62"	I	No	Yes

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
1A1	RC, RCP Motor 1A1, Elev. 57'	I	No	No
1A2	RC, RCP Motor 1A2, Elev. 57'	I	No	No
1B1	RC, RCP Motor 1B1, Elev. 57'	I	No	No
1B2	RC, RCP Motor 1B2, Elev. 57'	I	No	No
RC 005-34A	RC, Reactor Bldg, Elev. 68'	A	No	No
RC 005-34B	RC, Reactor Bldg, Elev. 68'	A	No	No
RC 005-36	RC, Reactor Bldg, Elev. 68'	A	No	No
RC 005-12B	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-12B	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-12A	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-55C	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-55B	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-62A	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-89	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-90	RC, Reactor Bldg, Elev. 80'	A	No	No
RC 005-98	RC, Reactor Bldg, Elev. 80'	A	No	No

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
MS 649-319	MS, Reactor Bldg, Elev. 82'	A	No	No
MS 548-5	MS, Reactor Bldg, Elev. 82'	A	No	No
MS 1076-3164	MS, M.S. Trestle, Elev. 62'	A	No	No
MS 649-314	MS, Reactor Bldg, Elev. 55'	I	No	No
MS 649-314	MS, Reactor Bldg, Elev. 55'	I	No	No
MS 649-310	MS, Reactor Bldg, Elev. 50'	I	No	No
MS 548-16A	MS, Reactor Bldg, Elev. 30'	A	No	Yes
MS 548-9	MS, Reactor Bldg, Elev. 50'	I	No	Yes
MS 548-9	MS, Reactor Bldg, Elev. 50'	I	No	Yes
BF 549-7	BF, Reactor Bldg, Elev. 40'	I	No	No
BF 549-7	BF, Reactor Bldg, Elev. 40'	I	No	No
BF 549-8	BF, Reactor Bldg, Elev. 40'	I	No	Yes
BF 549-11	BF, Reactor Bldg, Elev. 50'	I	No	No
BF 549-11	BF, Reactor Bldg, Elev. 50'	I	No	No
BF 549-17	BF, Reactor Bldg, Elev. 36'	A	No	Yes
BF 661-407	BF, Reactor Bldg, Elev. 40'	I	No	No
BF 661-407	BF, Reactor Bldg, Elev. 40'	I	No	No

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
BF 661-416	BF, Reactor Bldg, Elev, 50'	I	No	No
BF 661-416	BF, Reactor Bldg, Elev, 50'	I	No	No
BF 661-4020	BF, Reactor Bldg, Elev. 36'	A	No	No
SI 968-210	SI, Reactor Bldg, Elev. 16'	I	No	No
SI 968-565	SI, Reactor Bldg, Elev. 25'	A	No	No
SI 968-1205	SI, Reactor Bldg, Elev. 30'	A	No	No
SI 968-1207	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 969-1190	SI, Reactor Bldg, Elev. 20'	I	No	No
SI 969-1216	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 969-6193	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 969-6195	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 969-6198	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 969-6201	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 969-6217	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 969-6217	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 970-1210	SI, Reactor Bldg, Elev. 33'	I	No	No
SI 970-1248	SI, Reactor Bldg, Elev. 20'	I	No	No

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
SI 970-1251	SI, Reactor Bldg, Elev. 20'	A	No	No
SI 971-6	SI, Reactor Bldg, Elev. 20'	I	No	No
SI 971-1229	SI, Reactor Bldg, Elev. 20'	I	No	No
SI 971-6229	SI, Reactor Bldg, Elev. 20'	I	No	No
SI 971-6236	SI, Reactor Bldg, Elev. 20'	I	No	No
SI 972-1243	SI, Reactor Bldg, Elev. 25'	A	No	No
SI 972-6240	SI, Reactor Bldg, Elev. 16'	I	No	No
SI 973-240	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 973-6219	SI, Reactor Bldg, Elev. 36'	I	No	No
SI 973-6224	SI, Reactor Bldg, Elev. 18'	A	No	No
SI 868-64	SI, RAB, Elev. 4'	A	No	No
SI 868-111	SI, RAB, Elev. 4'	A	No	No
SI 868-163	SI, RAB, Elev. 4'	A	No	No
SI 868-410	SI, RAB, Elev. 4'	A	No	No
SI 676-67	SI, RAB, Elev. 4'	A	No	No
SI 676-67	SI, RAB, Elev. 4'	A	No	No
SI 676-105	SI, RAB, Elev. 4'	A	No	No

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
SI 676-105	SI, RAB, Elev. 4'	A	No	No
SI 676-127	SI, RAB, Elev. 4'	A	No	No
SI 676-129	SI, RAB, Elev. 4'	A	No	No
SI 676-250	SI, RAB, Elev. 24'	A	No	No
SI 676-2475	SI, RAB, Elev. 30'	A	No	No
SI 676-2475A	SI, RAB, Elev. 30'	A	No	No
SI 676-4505	SI, RAB, Elev. 7'	A	No	No
SI-V-1	SI, RAB, Elev. 4'	A	No	No
SI-V-1	SI, RAB, Elev. 4'	A	No	No
SPS-417	Pressurizer Spray, Reactor Bldg, Elev, 50'	I	No	No
SPS-27	Pressurizer Spray, Reactor Bldg, Elev,. 50'	I	No	No
SPS 467	Pressurizer Spray, Reactor Bldg, Elev. 80'	A	No	No
SPS-777	Pressurizer Spray, Reactor Bldg, Elev. 80'	A	No	No
CS-832-118	CS, Reactor Bldg, Elev. 125'	A	No	Yes

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
CS-878-115	CS, Reactor Bldg, Elev. 18'	A	No	No
CC-1865-9	CC, Reactor Bldg, Elev. 25'	A	No	No
CC-1899-48	CC, Reactor Bldg, Elev. 25'	A	No	No
CC-1899-2208	CC, Reactor Bldg, Elev. 59'	A	No	No
CC-1852-6241	CC, Reactor Bldg, Elev. 25'	A	No	No
CC-1865-2207	CC, Reactor Bldg, Elev. 59'	A	No	No
CC-17-1	CC, RAB, Elev. 20'	A	No	No
CC-14-2	CC, RAB, Elev. 26'	A	No	No
CC-21-1	CC, RAB, Elev. 20'	A	No	No
CC-21-5	CC, RAB, Elev. 26'	A	No	No
CC-23-2	CC, RAB, Elev. 26'	A	No	No
CH-3-40	CH, RAB, Elev. 34'	A	No	No
CH-3-54	CH, RAB, Elev. 21'	A	No	No
CH-3-54A	CH, RAB, Elev. 21'	A	No	No
CH-3-75	CH, RAB, Elev. 23'	A	No	No

TABLE 3.7-2

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION	ACCESSIBLE OR INACCESSIBLE (A or I)	HIGH RADIATION ZONE (Yes or No)	ESPECIALLY DIFFICULT TO REMOVE (Yes or No)
MS-649-313	MS, Reactor Bldg, Elev. 80'	I	No	No
MS-649-313	MS, Reactor Bldg, Elev. 80'	I	No	No

*Snubbers may be added to safety related systems without prior License Amendment to Table 3.7-2 provided that safety evaluations, documentation and reporting are provided in accordance with 10 CFR 50.59 and that a revision to Table 3.7-2 is included with a subsequent License Amendment request.

TABLE 4.7-3

HYDRAULIC SNUBBER INSPECTION SCHEDULE

NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*	NEXT REQUIRED INSPECTION INTERVAL**
0	18 months $\pm$ 25%
1	12 months $\pm$ 25%
2	6 months $\pm$ 25%
3 or 4	124 days $\pm$ 25%
5, 6, or 7	62 days $\pm$ 25%
≥ 8	31 days $\pm$ 25%

* Snubbers may be categorized into two groups, "accessible" and "inaccessible". This categorization shall be based upon the snubber's accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

** The required inspection interval shall not be lengthened more than one step at a time.

D.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.4 As a minimum, the following D.C. electrical equipment and bus shall be energized and OPERABLE:

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

1 - 125-volt battery bank and charger supplying the above D.C. bus.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above complement of D.C. equipment and bus OPERABLE, establish CONTAINMENT INTEGRITY within 8 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.4.1 The above required 125-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying indicated power availability.

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

SURVEILLANCE REQUIREMENTS (Continued)

2. The pilot cell specific gravity, corrected to 77°F, is $\geq$ 1.20.

3. The pilot cell voltage is $\geq$ 2.084 volts.

4. The overall battery voltage is $\geq$ 125 volts.

b. At least once per 92 days by verifying that:

1. The voltage of each connected cell is $\geq$ 2.084 volts under float charge and has not decreased more than 0.14 volts from the value observed during the original acceptance test.

2. The specific gravity, corrected to 77°F, of each connected cell is $\geq$ 1.20 and has not decreased more than 0.01 from the average of the connected cells at the time of inspection.

3. The electrolyte level of each connected cell is between the minimum and maximum level indication marks.

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

d. At least once per 18 months, during shutdown, by verifying that the battery capacity, with the charger disconnected, is adequate to either:

   1. Supply and maintain in OPERABLE status all of the actual emergency loads for at least 2 hours when the battery is subjected to a battery service test, or

   2. Supply a dummy load of the following profile for at least 2 hours while maintaining the battery terminal voltage $\geq$ 100 volts:

      a) $\geq$ 773 amperes during the initial 40 seconds of the test,

      b) $\geq$ 570 amperes during the remainder of the first hour of the test, and

      c) $\geq$ 145 amperes during the remainder of the 2 hours.

At the completion of this battery test, the battery charger shall be demonstrated capable of recharging its battery at a rate of < 155 amperes while supplying normal D.C. loads. The battery shall be charged to at least 95% capacity in $\leq$ 24 hours.

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 shall be performed subsequent to the satisfactory completion of the required battery service test.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. electrical busses shall be OPERABLE and energized from sources of power other than a diesel generator set but aligned to an OPERABLE diesel generator set:

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Bus

3 - 480 volt Emergency MCC Busses

2 - 120 volt A.C. Instrument Busses

APPLICABILITY: MODES 5 and 6

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying indicated power availability.

D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

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

TRAIN "A" consisting of 125-volt D.C. bus No. 1A, 125-volt D.C. battery bank No. 1A and a full capacity charger.

TRAIN "B" consisting of 125-volt D.C. bus No. 1B, 125-volt D.C. battery bank No. 1B, and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 125-volt D.C. bus inoperable, restore the inoperable bus 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 a 125-volt D.C. battery and/or an associated charger inoperable, restore the inoperable battery and/or charger to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.3.1 Each D.C. bus train shall be determined OPERABLE and energized at least once per 7 days by verifying indicated power availability.

4.8.2.3.2 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The electrolyte level of each pilot cell is between the minimum and maximum level indication marks.

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 set with:

   1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

   2. A fuel storage system containing a minimum of 16,450 gallons of fuel, and

   3. A fuel transfer pump.

APPLICABILITY: MODES 5 and 6.

ACTION:

With less than the above minimum required A.C. electrical power sources OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until the minimum required A.C. electrical power sources are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

4.8.1.2 The above required A.C. electrical power sources shall be demonstrated OPERABLE by the performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2 except for requirement 4.8.1.1.2a.5.

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. electrical busses shall be OPERABLE and energized from sources of power other than the diesel generator sets:

4160	volt Emergency Bus	1A3
4160	volt Emergency Bus	1B3
480	volt Emergency Bus	1A2
480	volt Emergency Bus	1B2
480	volt Emergency MCC Busses	1A5, 1A6, 1A7
480	volt Emergency MCC Busses	1B5, 1B6, 1B7
120	volt A.C. Instrument Bus	1MA
120	volt A.C. Instrument Bus	1MB
120	volt A.C. Instrument Bus	1MC
120	volt A.C. Instrument Bus	IMD

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With less than the above complement of A.C. busses OPERABLE, restore the inoperable bus to OPERABLE status within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.1 The specified A.C. busses shall be determined OPERABLE and energized from A.C. sources other than the diesel generators at least once per 7 days by verifying indicated power availability.

SURVEILLANCE REQUIREMENTS (Continued)

voltage of 2500 volts. The megger test voltage shall be applied for at least one minute or until the reading remains steady for at least 15 seconds. The conductor to ground isolation resistance shall be verified to be at least 100 megohms.

2. D.C. proof testing at $\geq$ 25,000 volts each of the three installed spare 5000 volt cables (one each in the ducts between the switchgear and 1) the diesel generators, 2) the component cooling water pump motors, and 3) the intake cooling water pump motors) and verifying that for each of the cables, the measured leakage current, when monitored at nominal 30 second intervals for at least 10 minutes, does not increase after an initial current decrease and stabilization. If any one of the installed spare 5000 volt cables fail the D.C. proof test, the following actions shall be performed prior to increasing the RCS T_{avg} above 200°F:

   a) An inservice cable in the same specific category shall be disconnected, designated as the replacement spare cable, and subjected to the same D.C. proof test. If the results of testing this cable are acceptable, this cable shall be retained as the installed spare while the original spare cable shall be removed and replaced. The newly installed cable shall be connected and placed inservice as the replacement for the newly designated spare. However, if the results of testing this cable are unsatisfactory, all Class 1E 5000 volt cables in the same specific duct run category shall be D.C. proof tested at $\geq$ 25,000 volts.

   b) The cause of the failure(s) in the cables(s) shall be determined and reported to the Commission for evaluation and acceptability. If the failure was either caused by the cable's operating environment or if it was a generic type failure, all Class 1E 5000 volt underground cable installations shall be improved to a level that will significantly reduce the factor(s) identified to be the cause of failure, or all Class 1E 5000 volt underground cables shall be replace by a type demonstrated to be acceptable.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by megger testing a sample of the 600 volt and lower voltage Class 1E underground cables at a minimum test voltage of 1000 volts and verifying a minimum conductor to ground isolation resistance of 25 megohms with all conductors in the control cable assemblies, except the one under test, grounded. The megger test voltage shall be applied for at least one minute or until the reading remains steady for at least 15 seconds. The cables selected for megger testing shall include at least one of each type of cable (cables shall be categorized according to construction and materials used in fabrication) in each of the ducts between the switchgear and 1) the diesel generators, 2) the component cooling water pump motors, and 3) the intake cooling water pump motors. The cables selected for megger testing shall be selected on a rotating basis.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator set shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying the fuel level in the engine-mounted fuel tank.

2. Verifying the fuel level in the fuel storage tanks.

3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the engine-mounted tank.

4. Verifying the diesels start from ambient condition.

5. Verifying the generator is synchronized, loaded to $\geq$ 1300 kw, and operates for $\geq$ 60 minutes.

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

b. At least once per 31 days by verifying that a sample of diesel fuel from the fuel storage tank is within the acceptable limits specified in Table 1 of ASTM D975-68 when checked for viscosity, water and sediment.

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

1. Subjecting the diesels 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 $\geq$ 600 hp without tripping.

3. Simulating a loss of offsite power in conjunction with a safety injection actuation signal, and:

a) Verifying de-energization of the emergency busses and load shedding from the emergency busses.

b) Verifying the diesels start from ambient condition on the auto-start signal, energize the emergency busses with permanently connected loads, energize

SURVEILLANCE REQUIREMENTS (Continued)

the auto-connected emergency loads through the load sequencing system and operate for $\geq$ 5 minutes while the generator is loaded with the emergency loads.

c) Verifying that on the safety injection actuation signal, all diesel generator trips, except engine overspeed and generator differential, are automatically bypassed.

4. Verifying the diesel generator set operates for $\geq$ 60 minutes while loaded to $\geq$ 3500 kw.

5. Verifying that the auto-connected loads to each diesel generator set do not exceed the 2000 hour rating of 3730 kw.

6. Verifying that the automatic sequence timers are OPERABLE with each load sequence time within $\pm$ 10% of its required value.

d. At least once per 18 months by verifying that each fuel transfer pump transfers fuel from each fuel storage tank to the engine mounted fuel tanks on each diesel via the installed cross connection lines.

4.8.1.1.3 The Class 1E underground cable system shall be demonstrated OPERABLE:

a. 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.

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

1. Selecting on a rotating basis at least 3 (one each in the ducts between the diesel generators and the switchgear, between the switchgear and the component cooling water pump motors, and between the switchgear and the intake cooling water pump motors) Class 1E 5000 volt underground cables and megger testing the selected cables at a minimum test

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 generator sets each with:

   1. Engine-mounted fuel tanks containing a minimum of 152 gallons of fuel,

   2. A separate fuel storage system containing a minimum of 16,450 gallons of fuel, and

   3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

ACTION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE at least once per 24 hours by verifying correct breaker alignments and indicated power availability.

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling cavity shall be maintained uniform and sufficient to ensure that the more restrictive of following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of $\geq$ 1720 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 1720 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length CEA in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the refueling cavity shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

*The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two wide range logarithmic neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each wide range logarithmic neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days.

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for a minimum of 72 hours.

APPLICABILITY: During movement of irradicated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

CONTAINMENT PENETRATIONS

LIMITING CONDITION FOR OPERATION

3.9.4 The containment 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, except as provided in Table 3.6-2 of Specification 3.6.3.1, 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. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.4 Each of the above required containment 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 by:

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

b. Testing the containment isolation valves per the applicable portions of Specifications 4.6.3.1.1 and 4.6.3.1.2.

COMMUNICATIONS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: During CORE ALTERATIONS.

ACTION:

When direct communications between the control room and personnel at the refueling station cannot be maintained, suspend all CORE ALTERATIONS. The provisions fo Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

MANIPULATOR CRANE OPERABILITY

LIMITING CONDITION FOR OPERATION

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

a. A minimum capacity of 2000 pounds, and

b. An overload cut off limit of $\leq$ 3000 pounds.

APPLICABILITY: During movement of CEAs or fuel assemblies 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. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 The manipulator crane used for movement of CEAs or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 72 hours prior to the start of such operations by performing a load test of at least 2500 pounds and demonstrating an automatic load cut off when the crane load exceeds 3000 pounds.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2000 pounds shall be prohibited from travel over irradiated fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies in the storage pool.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.9.7 Crane interlocks and physical stops which prevent crane travel with loads in excess of 2000 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.

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 At least one shutdown cooling loop shall be in operation.

APPLICABILITY: MODE 6.

a. With less than one shutdown cooling loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 A shutdown cooling loop shall be determined to be in operation and circulating reactor coolant at a flow rate of $\geq$ 3000 gpm at least once per 24 hours.

CONTAINMENT ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the containment isolation system inoperable, close each of the penetrations providing direct access from the containment atmosphere to the outside atmosphere. The provisions of Specification 3.0.3 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 signal from two of the containment radiation monitoring instrumentation channels.

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated within the reactor pressure vessel.

APPLICABILITY: During movement of fuel assemblies or CEAs within the reactor pressure vessel while in MODE 6.

ACTION:

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

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.

REFUELING OPERATIONS

STORAGE POOL WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11 As a minimum, 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

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

ACTION:

With the requirement of the specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the fuel storage areas and restore the water level to within its limit within 4 hours. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11 The water level in the storage pool shall be determined to be at least its minimum required depth at least once per 7 days when irradiated fuel assemblies are in the fuel storage pool.

FUEL POOL VENTILATION SYSTEM - FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.12 At least one fuel pool ventilation system shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the spent fuel pool.

ACTION:

a. With no fuel pool ventilation system OPERABLE, suspend all operations involving movement of fuel within the spent fuel pool or crane operation with loads over the spent fuel pool until at least one fuel pool ventilation system is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required fuel pool ventilation system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating flow through the HEPA filter and charcoal adsorber train and verifying that the train operates for at least 15 minutes.

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

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the charcoal adsorbers remove $\geq$ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm $\pm$ 10%.

2. Verifying that the HEPA filter banks remove $\geq$ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm $\pm$ 10%.

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 70% for radioactive elemental iodide when the sample is tested in accordance with ANSI N510-1975 (130°C, 95% R.H.). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 10,350 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by:

   1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 4.15 inches Water Gauge while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

   2. Verifying that the air flow distribution is uniform within 20% across HEPA filters and charcoal adsorbers when tested in accordance with ANSI N510-1975.

   3. Verifying that the ventilation system maintains the spent fuel storage pool area at a negative pressure of ≥ 1/8 inches Water Gauge relative to the outside atmosphere during system operation.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

SPENT FUEL CASK CRANE

LIMITING CONDITION FOR OPERATION

3.9.13 The maximum load which may be handled by the spent fuel cask crane shall not exceed 25 tons.

APPLICABILITY: Whenever irradiated fuel assemblies are in the 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

4.9.13 The loaded weight of a spent fuel assembly cask shall be verified to not exceed 25 tons prior to attaching it to the spent fuel cask crane.

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of CEA worth and shutdown margin provided:

a. Reactivity equivalent to at least the highest estimated CEA worth is available for trip insertion from OPERABLE CEA(s), and

b. All part length CEAs are at least 90% withdrawn and OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

a. With the reactor critical ($K_{eff} \geq 1.0$) and with less than the above reactivity equivalent available for trip insertion or the part length CEAs not within their withdrawal limits, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length and part length CEA 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 OPERABLE by verifying its CEA drop time to be $\leq$ 3.3 seconds within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.1.3 The part length CEAs shall be demonstrated OPERABLE by moving each part length CEA $\geq$ 7.5 inches within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 and 3.2.3 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, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 and 3.2.3 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.

SERVEILLANCE 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, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 or 3.2.3 are suspended and shall be verified to be within the test power plateau.

4.10.2.2 The linear heat rate shall be determined to be within the limits of Specification 3.2.1 by monitoring it continuously with the Incore Detector Monitoring System pursuant to the requirements of Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.2.2 or 3.2.3 are suspended.

SPECIAL TEST EXCEPTIONS

PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.10.3 The minimum temperature and pressure conditions for reactor criticality of Specifications 3.1.1.5 and 3.4.9.1 may be suspended during low temperature PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5 percent of RATED THERMAL POWER,

b. The reactor trip setpoints on the OPERABLE power range neutron flux monitoring channels are set at $\leq$ 20% of RATED THERMAL POWER, and

c. The Reactor Coolant System temperature and pressure relationship is maintained within the acceptable region of operation shown on Figure 3.4-2.

APPLICABILITY: MODE 2.

ACTION:

a. With the THERMAL POWER > 5 percent of RATED THERMAL POWER, immediately open the reactor trip breakers.

b. With the Reactor Coolant System temperature and pressure relationship within the region of unacceptable operation on Figure 3.4-2, immediately open the reactor trip breakers and restore the temperature-pressure relationship to within its limit within 30 minutes; perform the analysis required by Specification 3.4.9.1 prior to the next reactor criticality.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The Reactor Coolant System shall be verified to be within the acceptable region for operation of Figure 3.4-2 at least once per hour.

4.10.3.2 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour.

4.10.3.3 Each wide range logarithmic and power level channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specification 3.4.1 may be suspended during the performance of 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 ≤ 20% of RATED THERMAL POWER.

APPLICABILITY: During PHYSICS TESTS and Thermal-Hydraulic Tests.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately trip the reactor.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be ≤ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 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 PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

CENTER CEA MISALIGNMENT

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 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 4.10.5.2 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/or 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 Specifications 4.2.1.3 and 3.3.3.2 during PHYSICS TESTS above 5% of RATED THERMAL POWER in which the requirements of Specifications 3.1.3.1 and/or 3.1.3.6 are suspended.

BASES

FOR

SECTIONS 3.0 AND 4.0

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

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 states the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.1 calls for each Reactor Coolant System safety injection tank to be OPERABLE and provides explicit ACTION requirements when one safety injection tank is inoperable. Under the terms of Specification 3.0.3, if more than one safety injection tank is inoperable, the facility is required to be in at least HOT STANDBY within 1 hour and in COLD SHUTDOWN within the following 30 hours.

3.0.4 This specification provides that entry into an OPERATIONAL MODE or other specified applicability condition must be made with (a) the full complement of required systems, equipment or components OPERABLE and (b) all other parameters as specified in the Limiting Conditions for Operation being met without regard for allowable deviations and out of service provisions contained in the ACTION statements.

The intent of this provision is to insure that facility operation is not initiated with either required equipment or systems inoperable or other specified limits being exceeded.

Exceptions to this provision have been provided for a limited number of specifications when startup with inoperable equipment would not affect plant safety. These exceptions are stated in the ACTION statements of the appropriate specifications.

BASES

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.

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 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 requirement 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.

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 2.45% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

3/4.1.1.3 BORON DILUTION AND ADDITION

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 changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 11,400 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limiting values assumed for the MTC used in the accident and transient analyses were $+ 0.5 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels $<$ 70% of RATED THERMAL POWER, $+ 0.2 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels $>$ 70% of RATED THERMAL and $- 2.5 \times 10^{-4}$ $\Delta k/k/°F$ at RATED THERMAL POWER. Therefore, these limiting values are included in this specification. Determination of MTC at the specified conditions ensures that the maximum positive and/or negative values of the MTC will not exceed the limiting values.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating 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 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 all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 3,478 gallons of 8.0% boric acid solution from the boric acid tanks or 59,000 gallons of 1720 ppm borated water from the refueling water tank.

The requirements for a minimum contained volume of 371,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

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 change in the event the single injection system becomes inoperable.

BASES

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability required below 200°F is based upon providing a 1% $\Delta k/k$ SHUTDOWN MARGIN at 140°F during refueling with all full and part length control rods withdrawn. This condition requires either 5,650 gallons of 8.0% boric acid solution from the boric acid tanks or 100,000 gallons of 1720 ppm borated water from the refueling water tank.

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 a CEA ejection accident 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 criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment ($\geq$ 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 (< 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

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. The reactor

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time of 3.3 seconds used in the accident analyses. Measurement with $T_{avg} \geq 515°F$ 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.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

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 and part 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.

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, provide 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 3.1.3.2, 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 3.2.3 are satisfied, and 4) the TOTAL 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 1.10, 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, 5) a THERMAL POWER measurement uncertainty factor of 1.02, and 6) a rod bow penalty factor of 1.05.

3/4.2.2 and 3/4.2.3 TOTAL RADIAL PEAKING FACTOR - F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_r^T and T_q are provided to ensure that the assumptions used in the analysis for establishing the DNB Margin, Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If either F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the

TOTAL RADIAL PEAKING FACTOR - F_r^T AND AZIMUTHAL POWER TILT - T_q (continued)

ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the DNB Margin, 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 value of T_q that must be used in the equation $F_r^T = F_r^P (1 + T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_r^T and T_q are within their limits provide assurance that the actual values of F_r^T and T_q do not exceed the assumed values. Verifying F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.4 FUEL RESIDENCE TIME

The limitation on fuel burnup during the initial fuel cycle ensures that fuel cladding collapse will not occur. Performance data from similar fuel rods and analyses of the installed fuel rods show that cladding collapse will not occur until well beyond the proposed first cycle of operation which is about 11,200-12,000 Effective Full Power Hours. However, operation beyond the first cycle will require further analyses.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

3/4.2.5 DNB PARAMETERS

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that hte 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.

BASES

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and bypasses ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination therof exceeds its setpoint, 2) the specified coincidence logic is maintained, 3) sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance, and 4) sufficient system functional capability is available for protective and ESF purposes from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests performed at the minimum frequencies are sufficient to demonstrate this capability.

The measurement of response time at the specified frequencies provides assurance that the protective and ESF action function associated with each channel is completed within the time limit assumed in the accident analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such tests demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served

RADIATION MONITORING INSTRUMENTATION (Continued)

by the individual channels and 2) an alarm is initiated when the radiation level alarm setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capbility 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.

3/4.3.3.4. METEOROLOGICAL INSTRUMENTATION

The OPERABILITY of the meteorological instrumentation ensures that sufficient meteorological data is available for estimating potential radiation doses to the public as a result of routine or accidental release of radioactive materials to the atmosphere. This capability is required to evaluate the need for initiating protective measures to protect the health and safety of the public and is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs", February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of HOT SHUTDOWN of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criteria 19 of 10 CFR 50.

3/4.3.3.6 CHLORINE DETECTION SYSTEMS

The operability of the chlorine detection systems ensures that an accidental chlorine release will be detected promptly and the necessary protective actions will be automatically initiated to provide protection for control room personnel. Upon detection of a high concentration of chlorine, the control room emergency ventilation system will automatically isolate the control coom and initiate its operation in the recirculation mode of operation to provide the required protection. The chlorine detection systems required by this specification are consistent with the recommendations of Regulatory Guide 1.95, "Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release", February 1975.

BASES

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. STARTUP and POWER OPERATION may be initiated and may proceed with one or two reactor coolant pumps not in operation after the setpoints for the Power Level-High, Reactor Coolant Flow-Low, and Thermal Margin/Low Pressure trips have been reduced to their specified values. Reducing these trip setpoints ensures that the DNBR will be maintained above 1.30 during three pump operation and that during two pump operation the core void fraction will be limited to ensure parallel channel flow stability within the core and thereby prevent premature DNB.

A single reactor coolant loop with its steam generator filled above the low level trip setpoint provides sufficient heat removal capability for core cooling while in MODES 2 and 3; however, single failure considerations require plant cooldown if component repairs and/or corrective actions cannot be made within the allowable out-of-service time.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 2×10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

SAFETY VALVES (Continued)

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, 1974 Edition.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer--Pressure-High signal, minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two steam generators capable of removing decay heat, combined with the requirements of Specifications 3.7.1.1, 3.7.1.2 and 3.7.1.3 ensures adequate decay heat removal capabilities for RCS temperatures greater than 300°F if one steam generator becomes inoperable due to single failure considerations. Below 300°F, decay heat is removed by the shutdown cooling system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

3/4.4.5 STEAM GENERATORS (Continued)

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameter limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameter 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 gallon per minute, total). 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 1 gallon per minute 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 the all volatile treatment (AVT) of secondary coolant. However, even if a defect of similar type should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required of all tubes with imperfections exceeding the plugging limit which, by the definition of Specification 4.4.5.4.a is 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 to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.6.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the Reactor Coolant Pressure Boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.6.2 REACTOR COOLANT SYSTEM LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 GPM. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 GPM IDENTIFIED LEAKAGE limitation provides allowance for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The 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.

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 reduce the potential for Reactor coolant System leakage or failure due to stress corrosion. Maintaining the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are

time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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 interim 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 NRC is finalizing site specific criteria which will be used as the basis for the reevaluation of the specific activity limits of this site. This reevaluation may result in higher limits.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 μCi/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 μCi/gram DOSE EQUIVALENT I131 but within the limits shown on Figure 3.4-1 must be restricted to no more than 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.

Reducing T_{avg} to < 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 5.2.1 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.

BASES

The heatup and cooldown limit curves (Figure 3.4-2) are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup or cooldown rates of up to 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 2.4-2.

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>1 Mev) irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, can be predicted using Figure B 3/4.4-1. The heatup and cooldown limit curves shown on Figure 3.4-2 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period, as well as adjustments for possible errors in the pressure and temperature sensing instruments.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel. The heatup and cooldown curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The pressure-temperature limit lines shown on Figure 3.4-2 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The maximum RT_{NDT} for all reactor coolant system pressure-retaining materials, with the exception of the reactor pressure vessel, has been determined to be 50°F. The Lowest Service Temperature limit line shown on Figure 3.4-2 is based upon this RT_{NDT} since Article NB-2332 (Summer Addenda of 1972) of Section III of the ASME Boiler and Pressure Vessel Code requires the Lowest Service Temperature to be RT_{NDT} + 100°F

TABLE B 3/4.4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	NDTT F	50 FT-LB/35* MIL TEMP F LONG	50 FT-LB/35* MIL TEMP F TRANS	RTNDT F	MIN. UPPER SHELF FT-LB ** LONG	MIN. UPPER SHELF FT-LB ** TRANS
Vessel Flange Forging	C-1-1	A508Cl.2	-	.008	+20	+30	-	-	128	-
Bottom Head Plate	C-10-1	A533BCl.1	-	.010	-40	+18	-	-	118	-
Bottom Head Plate	C-9-2	A533BCl.1	-	.011	-40	-26	-	-	145	-
Bottom Head Plate	C-9-3	A533BCl.1	-	.013	-70	-34	-	-	148	-
Bottom Head Plate	C-9-1	A533BCl.1	-	.011	-30	+2	-	-	135	-
Inlet Nozzle	C-4-3	A508Cl.2	-	.005	0	-23	-	-	109	-
Inlet Nozzle	C-4-2	A508Cl.2	-	.004	0	-5	-	-	140	-
Inlet Nozzle	C-4-1	A508Cl.2	-	.005	+10	-30	-	-	141	-
Inlet Nozzle	C-4-4	A508Cl.2	-	.004	0	-50	-	-	132	-
Inlet Nozzle Ext.	C-16-3	A508Cl.2	-	.001	+10	0	-	-	135	-
Inlet Nozzle Ext.	C-16-2	A508Cl.2	-	.011	+10	0	-	-	135	-
Inlet Nozzle Ext.	C-16-1	A508Cl.2	-	.011	+10	0	-	-	135	-
Inlet Nozzle Ext.	C-16-4	A508Cl.2	-	.011	+10	0	-	-	135	-

* Average Value from Curve
** Minimum Value at 100% Shear

TABLE B 3/4.4-1 (Cont'd)

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	NDTT F	50 FT-LB/35* MIL TEMP F LONG	50 FT-LB/35* MIL TEMP F TRANS	RTNDT F	MIN. UPPER SHELF FT-LB ** LONG	MIN. UPPER SHELF FT-LB ** TRANS
Outlet Nozzle	C-3-1	A508Cl.2	-	.009	+10	+50	-	-	118	-
Outlet Nozzle	C-3-2	A508Cl.2	-	.010	-20	+60	-	-	108	-
Outlet Nozzle Ext.	C-17-1	A508Cl.2	-	.013	+20	+27	-	-	126	-
Outlet Nozzle Ext.	C-17-2	A508Cl.2	-	.013	+20	+27	-	-	126	-
Upper Shell Plate	C-6-3	A533BCl.1	-	.011	-10	24	-	-	117	-
Upper Shell Plate	C-6-2	A533BCl.1	-	.010	-30	9	-	-	113	-
Upper Shell Plate	C-6-1	A533BCl.1	-	.012	+10	34	-	-	104	-
Inter. Shell Plate	C-7-1	A533BCl.1	0.11	0.004	0	+10	-	-	126	-
Inter. Shell Plate	C-7-2	A533BCl.1	0.11	0.004	-30	+10	-	-	126	-
Inter. Shell Plate	C-7-3	A533BCl.1	0.11	0.004	-30	+30	-	-	124	-
Lower Shell Plate	C-8-3	A533BCl.1	0.12	0.004	0	+13	-	-	135	-
Lower Shell Plate	C-8-1	A533BCl.1	0.15	0.006	-10	+16	-	-	126	-
Lower Shell Plate	C-8-2	A533BCl.1	0.15	0.006	0	+16	-	-	122	-
Closure Head Flange	C-2	A508Cl.2	-	.008	+10	-6	-	-	140	-
Closure Head Peels	C-21-2	A533BCl.2	-	.012	-30	+26	-	-	132	-
Closure Head Peels	C-21-2	A533BCl.1	-	.012	-30	+26	-	-	132	-

TABLE B 3/4.4-1 (Cont'd)

REACTOR VESSEL TOUGHNESS

COMPONENT	COMP CODE	MATERIAL TYPE	CU %	P %	NDTT F	50 FT-LB/35* MIL TEMP F LONG	50 FT-LB/35* MIL TEMP F TRANS	RTNDT F	MIN. UPPER SHELF FT-LB ** LONG	MIN. UPPER SHELF FT-LB ** TRANS
Closure Head Peels	C-21-1	A533BC1.1	-	.013	-10	-10	-	-	125	-
Closure Head Peels	C-21-1	A533BC1.1	-	.013	-10	-10	-	-	125	-
Closure Heal Peels	C-21-2	A533BC1.1	-	.012	-30	+26	-	-	132	-
Closure Head Peels	C-21-3	A533BC1.1	-	.013	-40	+16	-	-	121	-
Closure Head Dome	C-20-1	A533BC1.1	-	.014	-10	+37	-	-	101	-

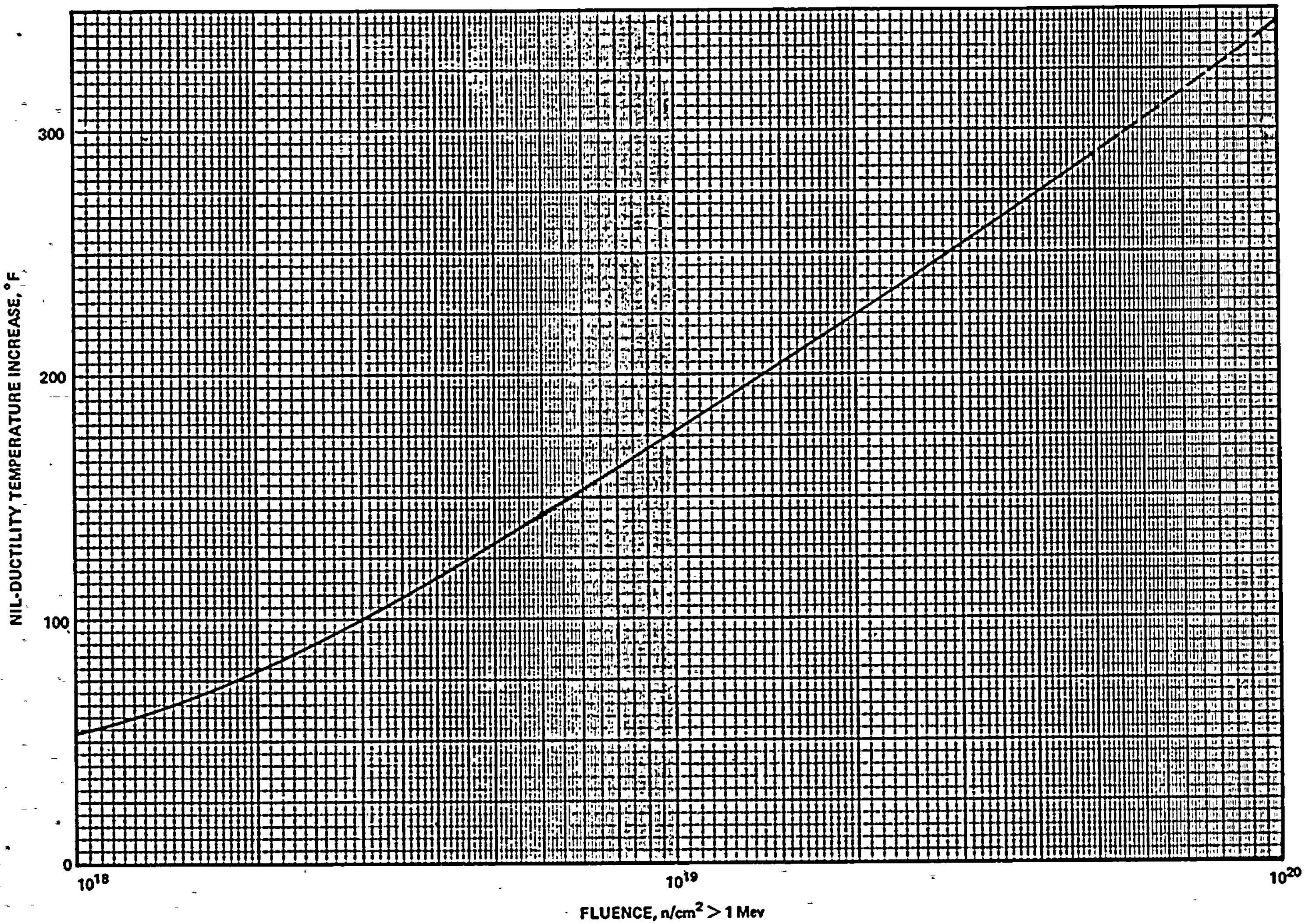


FIGURE B 3/4.4-1

Nil-Ductility Temperature Increase as a Function of Fast Neutron Fluence (E> 1mev)

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 number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-5 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

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.

3/4.4.10 STRUCTURAL INTEGRITY

The required inspection programs for the Reactor Coolant System components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for the Reactor Coolant System components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code "Inservice Inspection of Nuclear Reactor Coolant Systems", 1971 Edition, and Addenda through Winter 1972.

All areas scheduled for volumetric examination have been pre-service mapped using equipment, techniques and procedures anticipated for use during post-operation examinations. To assure that consideration is given to the use of new or improved inspection equipment, techniques and procedures, the Inservice Inspection Program will be periodically reviewed on a 5 year basis.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most reactor coolant loop components except the reactor vessel. The reactor vessel requires special consideration because of the radiation levels and the requirement for remote underwater accessibility.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle and dye penetrant testing of selected parts.

BASES

The nondestructive testing for repairs on components greater than 2 inches diameter gives a high degree of confidence in the integrity of the system, and will detect any significant defects in and near the new welds. Repairs on components 2 inches in diameter or smaller receive a surface examination which assures a similar standard of integrity. In each case, the leak test will ensure leak tightness during normal operation.

For normal opening and reclosing, the structural integrity of the Reactor Coolant System is unchanged. Therefore, satisfactory performance of a system leak test at 2235 psia following each opening and subsequent reclosing is acceptable demonstration of the system's structural integrity. These leak tests will be conducted within the pressure-temperature limitations for Inservice Leak and Hydrostatic Testing and Figure 3.4-2.

The Safety Class 2 and 3 components will be pressure tested at least once toward the end of each inspection interval (10 years). The Safety Class 2 components having a design temperature above 400°F will be pressure tested at not less than 125 percent of the system design pressure while those components having a design temperature of 400°F and below will be pressure tested at 110 percent of design pressure. The Safety Class 3 components will be pressure tested at the levels indicated in Specification 4.4.10.3b.

3/4.4.11 CORE BARREL MOVEMENT

This specification is provided to ensure early detection of excessive core barrel movement if it should occur. Core barrel movement will be detected by using four excore neutron detectors to obtain Amplitude Probability Distribution (APD) and Special Analysis (SA). Baseline core barrel movement Alert Levels and Action Levels at nominal THERMAL POWER levels of 20%, 50%, 80% and 100% of RATED THERMAL POWER will be determined during the reactor startup test program.

A modification to the required monitoring program may be justified by an analysis of the data obtained and by an examination of the affected parts during the plant shutdown at the end of the first fuel cycle.

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The limit of one hour for operation with an inoperable safety injection tank minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional safety injection tank which may result in unacceptable peak cladding temperatures.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

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 provides this protection by dissolving in the sump water and causing its final pH to be raised to ≥ 7.0.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained. The requirement to dissolve a representative sample of TSP in a sample of RWST 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 (RWT)

The OPERABILITY of the 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.

3/4.6.1 CONTAINMENT VESSEL

3/4.6.1.1 CONTAINMENT VESSEL INTEGRITY

CONTAINMENT VESSEL INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure, P_a (39.6 psig). As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ or $\leq 0.75\ L_t$ (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 leak rate given in Specifications 3.6.1.1 and 3.6.1.2. The limitations on the air locks allow entry and exit into and out of the containment during operation and ensure through the surveillance testing that air lock leakage will not become excessive through continuous usage.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structural is prevented from exceeding its design negative pressure differential with respect to the annulus atmosphere of 0.70 psi and 2) the containment peak pressure does not exceed the design pressure of 44 psig during steam line break accident conditions.

The maximum peak pressure obtained from a steam line break accident is 41.6 psig. The limit of 2.4 psig for initial positive containment pressure will limit the total pressure to 44.0 psig which is the design pressure and is consistent with the accident analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitation on containment air temperature ensures that the containment vessel temperature does not exceed the design temperature of 264°F during LOCA conditions. The containment temperature limit is consistent with the accident 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 41.6 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.

3/4.6.2 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 capbility will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the accident analyses.

3/4.6.2.2 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.

3/4.6.3 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment. Containment isolation within the time limits specified ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

3/4.6.4 COMBUSTIBLE GAS CONTROL

The OPERABILITY of the equipment and systems required for the detection and control of hydrogen gas ensures that this equipment will be available to maintain the hydrogen concentration within containment below its flammable limit during post-LOCA conditions. Either recombiner unit is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment.

The containment fan coolers are used in a secondary function to ensure adequate mixing of the containment atmosphere following a LOCA. This mixing action will prevent localized accumulations of hydrogen from exceeding the flammable limit.

3/4.6.5 VACUUM RELIEF VALVES

The OPERABILITY of the containment vessel to annulus vacuum relief valves ensures that they will open at a pressure differential of 2.25 $\pm$ 0.25 inches Water Gauge. This condition is necessary to prevent exceeding the containment design limit for internal pressure differential of 0.70 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. This requirement is necessary to meet the assumptions used in the accident analyses and limit the site boundary radiation doses to within the limits of 10 CFR 100 during LOCA conditions.

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 accident 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 within two minutes after a LOCA.

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within its design pressure of 1025 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Code, 1971 Edition and ASME Code for Pumps and Valves, Class II. The total relieving capacity for all valves on all of the steam lines is 11.91×10^6 lbs/hr which is 106.7 percent the total secondary steam flow of 11.17×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of 2 OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation

$$SP = \frac{(X) - (Y)(V)}{X} \times (106.5)$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

106.5	=	Power Level-High Trip Setpoint for two loop operation
X	=	Total relieving capacity of all safety valves per steam line in lbs/hour (5.95×10^6 lbs/hr.)
Y	=	Maximum relieving capacity of any one safety valve in lbs/hour (7.44×10^5 lbs/hr.)

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 300°F from normal operating conditions in the event of a total loss of off-site power.

Any two of the three auxiliary feedwater pumps have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to 300°F where the shutdown cooling system may be placed into operation for continued cooldown.

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 for cooldown of the Reactor Coolant System to less than 300°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. The dose calculations for an assumed steam line rupture include the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blowdown in the event of a steam line rupture. This restriction is required to 1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and 2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the surveillance requirements are consistent with the assumptions used in the accident analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

A test program will be conducted during approximately the first 6 months of operation after initial criticality to establish the appropriate limits on the secondary water chemistry parameters and to determine the appropriate frequencies for monitoring these parameters. The results of this test program will be submitted to the Commission for review. The Commission will then issue a revision to this specification specifying the limits on the chemistry parameters and the frequencies for monitoring these parameters.

The test program will include an analysis of the chemical constituent of the makeup water for the St. Lucie Plant. The analysis shall identify the various traces of ions which upon concentration may have the potential for inducement for stress corrosion in the steam generator tubing. The test program shall also evaluate the efficiency of the water treatment systems in the St. Lucie facility for removal of such ions and the potential for addition of other ions resulting from the treatment method. The test program shall analyze concentration phenomena and the concentration rates in the steam generator and the secondary water system and shall consider concentration in the recirculating cooling water system.

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

The limitation on steam generator pressure and temperature ensures that the pressure induced stresses in the steam generators do not exceed the maximum allowable fracture toughness stress limits. The limitations of 70°F and 200-psig are based on a steam generator RT_{NDT} of 50°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 vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.4 INTAKE COOLING WATER SYSTEM

The OPERABILITY of the intake cooling water system ensures that sufficient cooling capacity is available for continued operation of vital components and Engineered Safety Feature equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the accident analyses.

3/4.7.5 ULTIMATE HEAT SINK

The limitations on the ultimate heat sink level and temperature ensure that sufficient cooling capacity is available to either 1) provide normal cooldown of the facility, or 2) to mitigate the effects of accident conditions within acceptable limits.

The limitations on minimum water level and maximum temperature are based on providing an adequate cooling water supply to safety related equipment until cooling water can be supplied from Big Mud Creek.

3/4.7.6 FLOOD PROTECTION

The limitation on flood protection ensures that facility will be adequately protected from flooding.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

The OPERABILITY of the control room emergency ventilation system ensures that 1) the ambient air temperature does not exceed the allowable temperature for continuous duty rating for the equipment and instrumentation cooled by this system and 2) the control room will remain habitable

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM (Continued)

for operations personnel during and following all credible accident conditions. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent, and maintaining the chlorine concentration within acceptable limits during and following a chlorine accident. This limitation is consistent with the requirements of General Design Criteria 10 of Appendix "A", 10 CFR 50.

3/4.7.8 ECCS AREA VENTILATION SYSTEM

The OPERABILITY of the ECCS area ventilation system ensures that radioactive materials leaking from the ECCS equipment following a LOCA are filtered prior to reaching the environment. The operation of this system and the resultant effect on offsite dosage calculations was assumed in the accident analyses.

3/4.7.9 SEALED SOURCE CONTAMINATION

The limitations on sealed source removable contamination ensure that the total body or individual organ irradiation does not exceed allowable limits in the event of ingestion or inhalation of the probable leakage from the source material. The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. Quantities of interest to this specification which are exempt from the leakage testing are consistent with the criteria of 10 CFR Parts 30.11-20 and 70.19. Leakage from sources excluded from the requirements of this specification is not likely to represent more than one maximum permissible body burden for total body irradiation if the source material is inhaled or ingested.

3/4.7.10 HYDRAULIC SNUBBERS

The hydraulic 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. The only 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.

BASES

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment (only ethylene propylene compounds to date) is based upon maintaining a constant level of snubber protection. Therefore, the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during an inspection of these snubbers determines the time interval for the next required inspection of these snubbers. 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 further assurance of snubber reliability, a representative sample of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. These tests will include stroking of the snubbers to verify proper piston movement, lock-up and bleed. Observed failures of these sample snubbers will require functional testing of additional units. To minimize personnel exposures, snubbers installed in high radiation zones or in especially difficult to remove locations (as identified in Table 3.7-2) may be exempted from these functional testing requirements provided the OPERABILITY of these snubbers was demonstrated during functional testing at either the completion of their fabrication or at a subsequent date.

BASES

The OPERABILITY of the A.C. and D.C. power sources and associated distribution systems during operation ensures that sufficient power will be available to supply the safety related equipment required for 1) the safe shutdown of the facility and 2) the mitigation and control of accident conditions within the facility. The minimum specified independent and redundant A.C. and D.C. power sources and distribution systems satisfy the requirements of General Design Criteria 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the accident analyses and are based upon maintaining at least one of each of the onsite A.C. and D.C. power sources and associated distribution systems OPERABLE during accident conditions coincident with an assumed loss of offsite power and single failure of the other onsite A.C. source.

The OPERABILITY of the minimum specified A.C. and D.C. power sources and associated distribution systems during shutdown and refueling ensures that 1) the facility can be maintained in the shutdown or refueling condition for extended time periods and 2) sufficient instrumentation and control capability is available for monitoring and maintaining the facility status.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration (1720 ppm) 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 volumes having direct access to the reactor vessel. The limitation on K_{eff} of no greater than 0.95 is sufficient to prevent reactor criticality with all full length rods (shutdown and regulating) fully withdrawn.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the wide range logarithmic range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY ensure that a release of radioactive material within containment will be restricted from leakage to the environment. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during CORE ALTERATIONS.

3/4.9.6 MANIPULATOR CRANE OPERABILITY

The OPERABILITY requirements of the cranes used for movement of fuel assemblies ensures that: 1) each crane has sufficient load capacity to lift a fuel element, and 2) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel assembly and CEA over irradiated fuel assemblies ensures that no more than the contents of one fuel assembly will be ruptured in the event of a fuel handling accident. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 COOLANT CIRCULATION

The requirement that at least one shutdown cooling loop be in operation is consistent with the assumptions in the safety analysis of the boron dilution accident and prevents local variations in boron concentrations, thus, minimizing the effects of an inadvertent boron dilution. It also assures that 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.

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 WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.12 FUEL POOL VENTILATION SYSTEM-FUEL STORAGE

The limitations on the fuel handling building ventilation system ensures that all radioactive material released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorber prior to discharge to the atmosphere. The OPERABILITY of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.13 SPENT FUEL CASK CRANE

The maximum load which may be handled by the spent fuel cask crane is limited to a loaded single element cask which is equivalent to approximately 25 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 single element cask could cause leakage from the spent fuel pool in excess of the maximum makeup capability.

BASES

3/4.10.1 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of CEA worth is immediately available for reactivity control when tests are performed for CEAs worth measurement. This special test exception is required to permit the periodic verification of the actual versus predicted core reactivity condition occurring as a result of fuel burnup or fuel cycling operations.

3/4.10.2 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual CEAs to be positioned outside of their normal group heights and insertion limits during the performance of such PHYSICS TESTS as those required to 1) measure CEA worth and 2) determine the reactor stability index and damping factor under xenon oscillation conditions.

3/4.10.3 PRESSURE/TEMPERATURE LIMITATION - REACTOR CRITICALITY

This special test exception permits the reactor to be critical at less than or equal 5% of RATED THERMAL POWER during low temperature PHYSICS TESTS required to measure such parameters as CEA worth and SHUTDOWN MARGIN.

3/4.10.4 PHYSICS TESTS

This special test exception permits PHYSICS TESTS and Thermal-Hydraulic Tests to be performed at less than or equal to 5% of RATED THERMAL POWER and is required to verify the fundamental nuclear characteristics of the reactor core and related instrumentation and the natural circulation capability of the reactor coolant system at low THERMAL POWER levels. This special test exception also permits the performance of CEA drop testing, reactor coolant flow measurements and flow coastdown testing described in Chapter 14.0 of the FSAR, and other special plant testing authorized under the provisions of 10 CFR 50.59, with no reactor coolant pumps in operation.

3/4.10.5 CENTER CEA MISALIGNMENT

This special test exception permits the center CEA to be misaligned during PHYSICS TESTS required to determine the isothermal temperature coefficient and power coefficient.

SECTION 5.0

DESIGN FEATURES

5.0 DESIGN FEATURES

5.1 SITE

EXCLUSION AREA

5.1.1 The exclusion area is shown on Figure 5.1-1.

LOW POPULATION ZONE

5.1.2 The low population zone is shown on Figure 5.1-2.

FLOOD CONTROL

5.1.3 The flood control provisions (dunes and slope protection) shall be designed and maintained in accordance with the original design provisions contained in Section 2.4.2.2 of the FSAR.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The containment structure is comprised of a steel containment vessel, having the shape of a right circular cylinder with a hemispherical dome and ellipisoidal bottom, surrounded by a reinforced concrete shield building. The radius of the shield building is at least 4 feet greater than the radius of circular cylinder portion of the containment vessel at any point.

5.2.1.1 CONTAINMENT VESSEL

a. Nominal inside diameter = 140 feet.

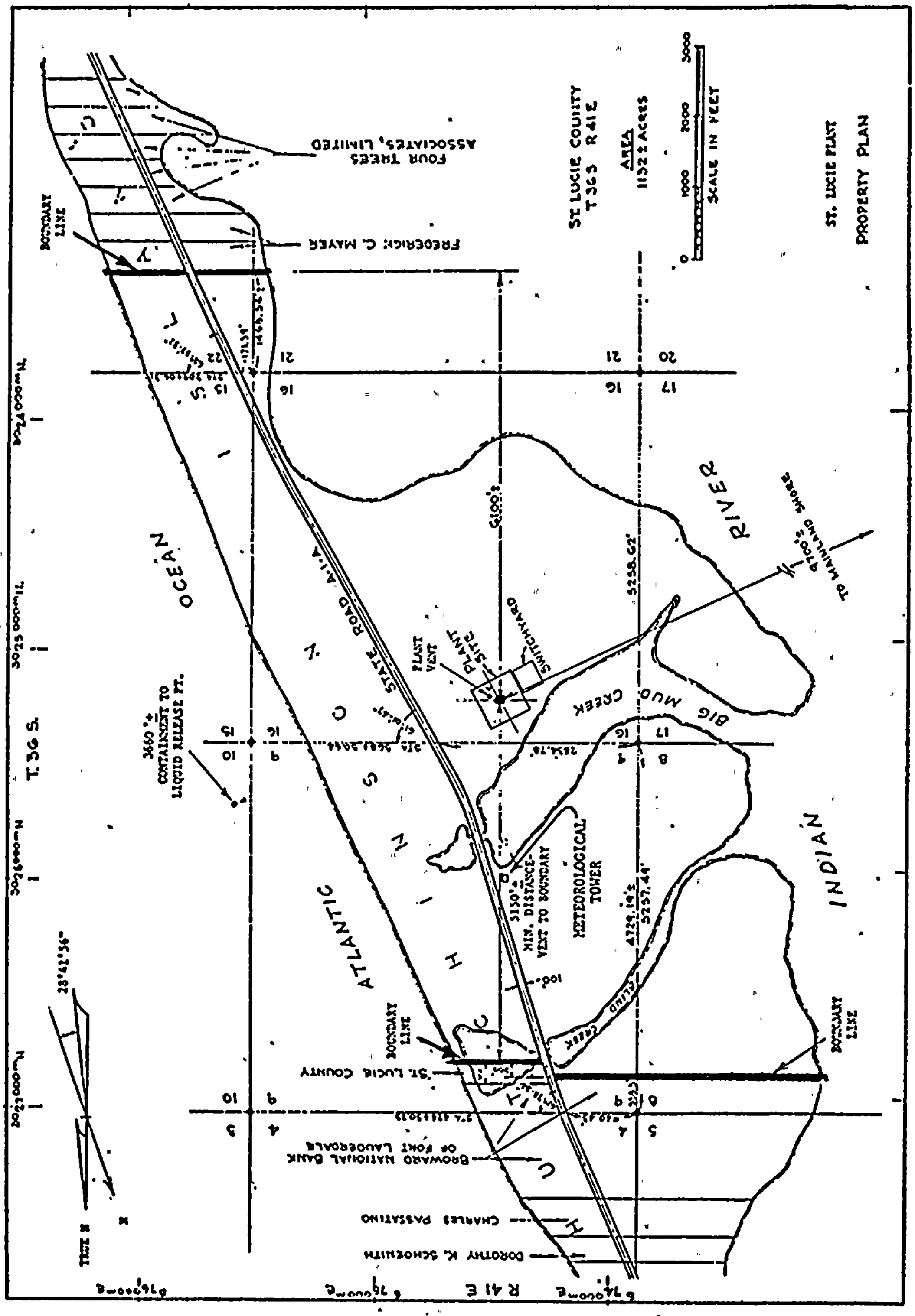
b. Nominal inside height = 232 feet.

c. Net free volume = 2.5×10^6 cubic feet.

d. Nominal thickness of vessel walls = 2 inches.

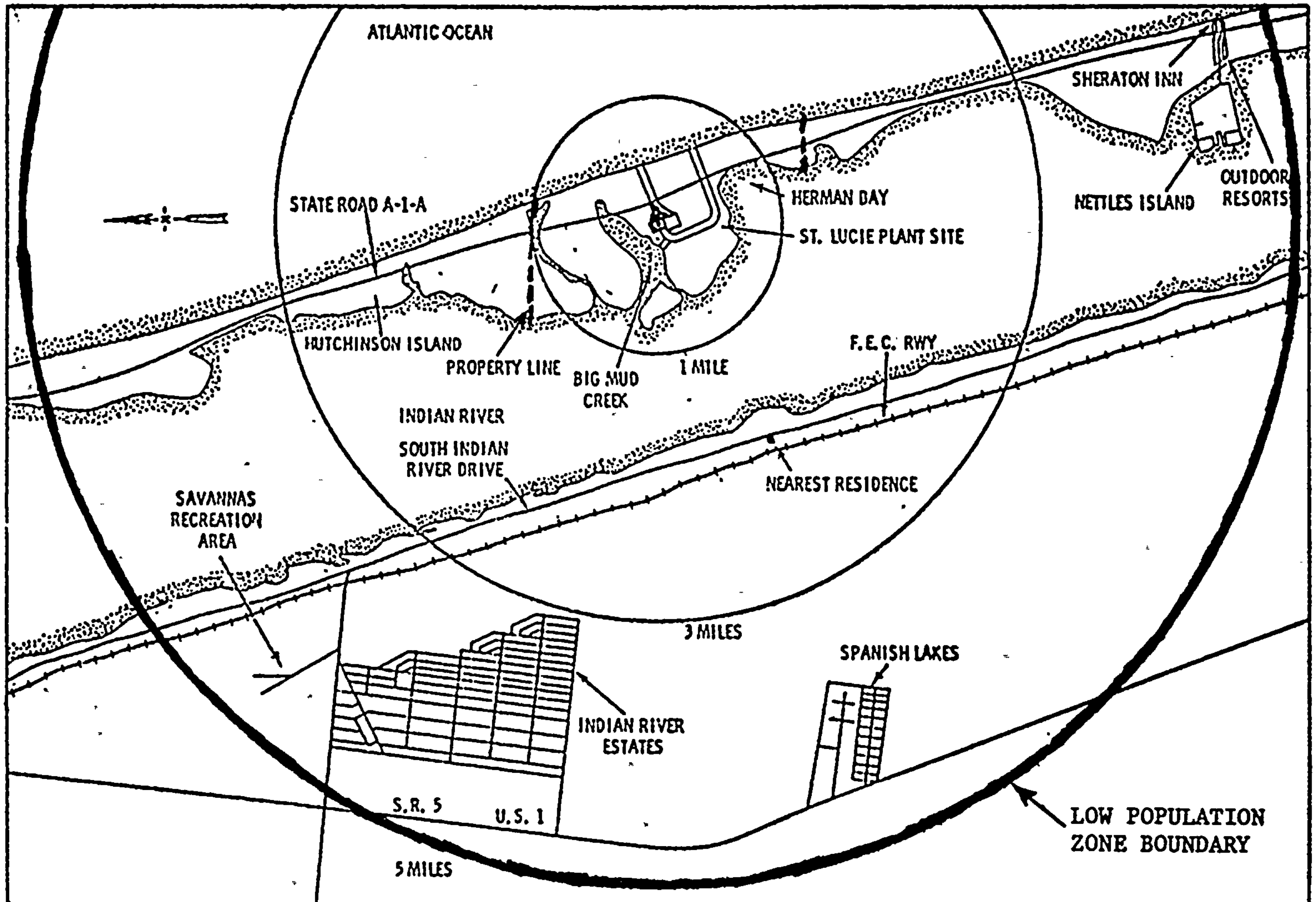
e. Nominal thickness of vessel dome = 1 inch.

f. Nominal thickness of vessel bottom = 2 inches.



EXCLUSION AREA

FIGURE 5.1-1



LOW POPULATION ZONE

FIGURE 5.1-2

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 base to the top of the dome) = 230.5 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 containment vessel is designed and shall be maintained for a maximum internal pressure of 44 psig and a temperature of 264°F.

PENETRATIONS

5.2.3 Penetrations through the containment structure are designed and shall be maintained in accordance with the original design provisions contained in Sections 3.8.2.1.10 and 6.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 217 fuel assemblies with each fuel assembly containing a maximum of 176 fuel rods clad with Zircoloy-4. Each fuel rod shall have a nominal active fuel length of 136.7 inches and contain a maximum total weight of 2250 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have maximum enrichment of 3.1 weight percent U-235.

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and 8 part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,100 $\pm$ 180 cubic feet at a nominal T_{avg} of 567°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with a nominal center-to-center distance of 21 inches for new fuel assemblies and 18 inches for spent fuel assemblies placed in the

CRITICALITY (Continued)

storage racks to ensure a K_{eff} equivalent to $\leq$ 0.95 with the storage pool filled with unborated water. The K_{eff} of $\leq$ 0.95 includes the conservative assumptions as described in Section 9.1 of the FSAR.

DRAINAGE

5.6.2 The fuel 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 pool is designed and shall be maintained with a storage capacity limited to no more than 304 fuel assemblies of which the 45 fuel assemblies in the 5 x 5 array and 5 x 4 array nearest the fuel cask compartment shall have decayed for at least 1000 hours.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as seismic Class I in Section 3.2.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 3.7 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower location shall be as shown on Figure 5.1-1.

5.9 COMPONENT CYCLE OR TRANSIENT LIMITS

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

TABLE 5.9-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMITS	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	40 Cycles of loss of load without immediate reactor trip	100% to 0% RATED THERMAL POWER
	40 cycles of loss of offsite A.C. electrical power	100% to 0% RATED THERMAL POWER
	400 reactor trips	100% to 0% RATED THERMAL POWER
	16 inadvertent auxiliary spray cycles	Spray line 650°F to 120°F in 1.5 seconds
	200 leak tests	Pressure $\geq$ 2235 psig
	10 hydrostatic pressure tests	Pressure $\geq$ 3110 psig
Secondary System	5 steam line breaks	Complete loss of secondary pressure
	200 leak tests	Pressure $\geq$ 985 psig
	10 hydrostatic pressure tests	Pressure $\geq$ 1235 psig

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Plant Manager shall be responsible for overall facility operation and shall delegate in writing the succession to this responsibility during his absence.

6.2 ORGANIZATION

OFFSITE

6.2.1 The offsite organization for facility management and technical support shall be as shown on Figure 6.2-1.

FACILITY STAFF

6.2.2 The Facility organization shall be as shown on Figure 6.2-2 and:

a. Each on duty shift shall be composed of at least the minimum shift crew composition shown in Table 6.2-1.

b. At least one licensed Operator shall be in the control room when fuel is in the reactor.

c. At least two licensed Operators shall be present in the control room during reactor start-up, scheduled reactor shutdown and during recovery from reactor trips.

d. An individual qualified in radiation protection procedures shall be on site when fuel is in the reactor.

e. All CORE ALTERATIONS after the initial fuel loading shall be directly supervised by either a licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling who has no other concurrent responsibilities during this operation.

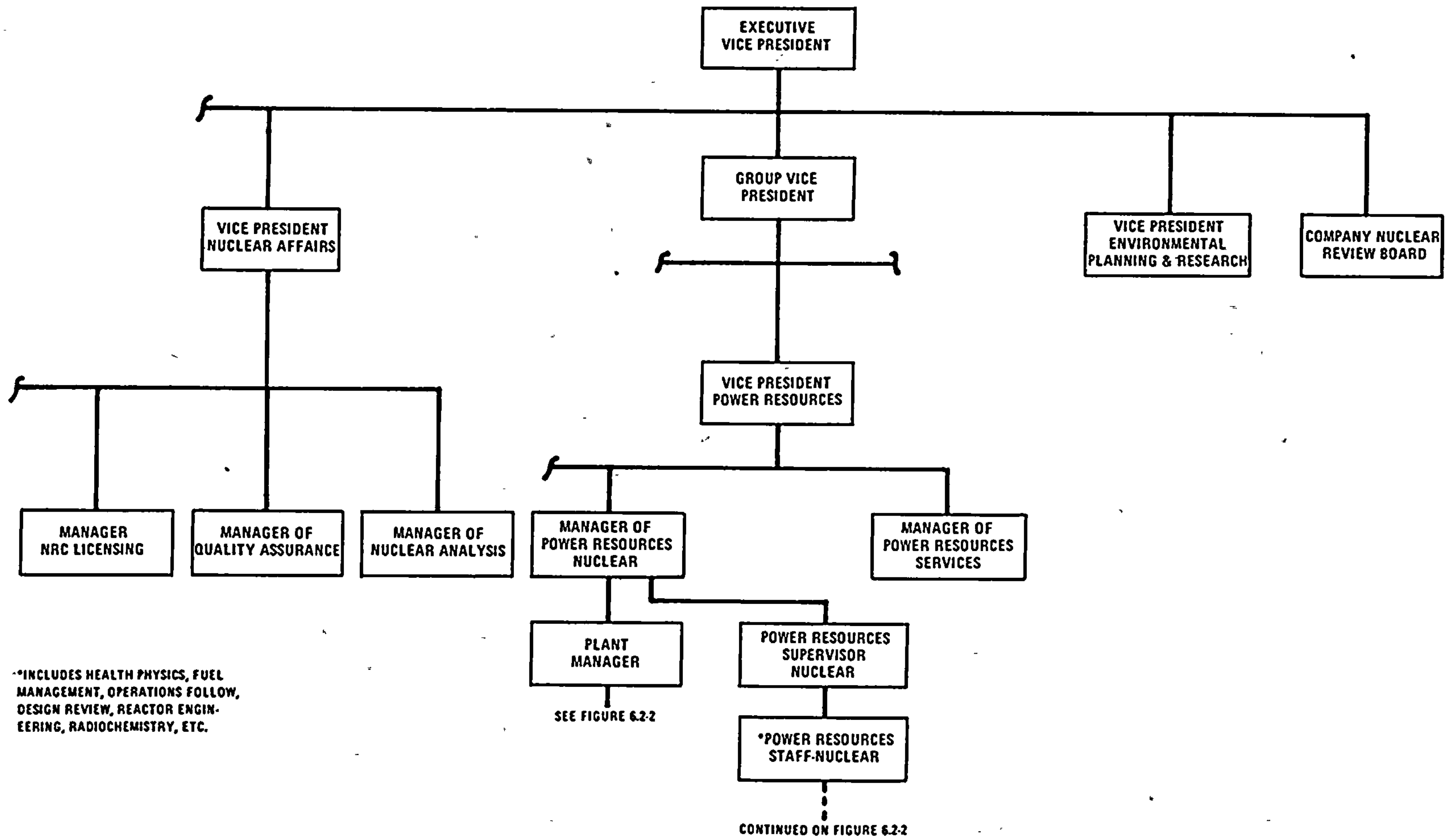


Figure 6.2-1 Offsite Organization for Facility Management and Technical Support

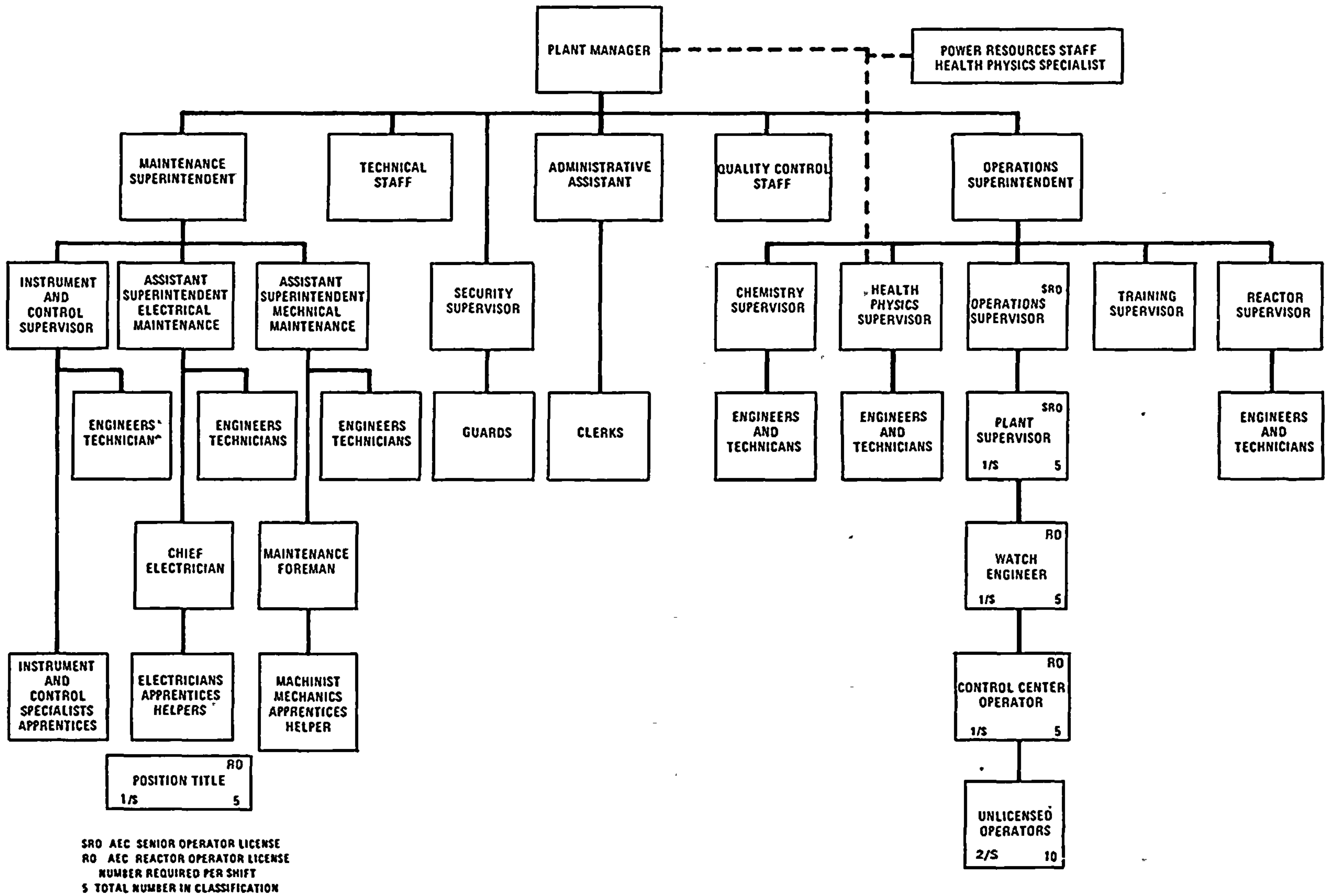


Figure 6.2-2 Facility Organization — St. Lucie Plant, Unit 1

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3, & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	2	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling, supervising CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less than the minimum requirements for a period of time not to exceed 2 hours in order to accomodate 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.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Radiation Protection Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility 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 N18.1-1971 and Appendix "A" of 10 CFR Part 55.

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
Member:	Instrument & Control Supervisor
Member:	Reactor Supervisor
Member:	Health Physics Supervisor
Member:	Technical Supervisor
Member:	Chemistry Supervisor
Member:	Quality Control Supervisor
Member:	Assistant Plant Supt. Mechanical
Member:	Assistant Plant Supt. Electrical

The FRG Chairman 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.

QUORUM

6.5.1.5 A quorum of the FRG shall consist of the Chairman 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) 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 plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Manager of Power Resources Nuclear, the Vice President of Power Resources and to the Chairman of the Company Nuclear Review Board.

f. Review of those REPORTABLE OCCURRENCES requires 24 hour notification to the Commission.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews and investigations and reports thereon as requested by the Chairman of the Company Nuclear Review Board.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Company Nuclear Review Board.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Company Nuclear Review Board.

AUTHORITY

6.5.1.7 The Facility Review Group shall:

a. Recommend to the Plant Manager written approval or disapproval of items considered under 6.5.1.6(a) through (d) above.

b. Render determinations in writing with regard to whether or not each item considered under 6.5.1.6(a) through (e) above constitutes an unreviewed safety question.

c. Provide written notification within 24 hours to the Vice President of Power Resources 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 6.1.1 above.

RECORDS

6.5.1.8 The Facility Review Group shall maintain written minutes of each meeting and copies shall be provided to the Vice President of Power Resources and 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

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:

Member:	Vice President of Nuclear Affairs
Member:	Chief Engineer Power Plants
Member:	Vice President of Power Resources
Member:	Power Plant Engineering Supervisor
Member:	Manager of Power Resources - Nuclear
Member:	Manager of QA
Member:	Power Plant Engineering Supervisor

The CNRB Chairman 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 facility operation following fuel loading and at least once per six months thereafter.

QUORUM

6.5.2.6 A quorum of CNRB shall consist of the Chairman or his designated alternate and four members including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

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 in Technical Specifications or licenses.

e. Violations of applicable statutes, 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 plant equipment that affect nuclear safety.

g. REPORTABLE OCCURRENCES requiring 24 hour notification to the Commission.

h. Any indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the Facility Review Group.

AUDITS

6.5.2.8 Audits of facility activities shall be performed under the cognizance of the CNRB. These audits shall encompass:

a. The conformance of facility operation to all 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 facility staff at least once per 12 months.

c. The results of all actions taken to correct deficiencies occurring in facility equipment, structures, systems or method of operation that affect nuclear safety at least once per 6 months.

d. The performance of all activities required by the Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The Facility Emergency Plan and implementing procedures at least once per 24 months.

f. The Facility Security Plan and implementing procedures at least once per 24 months.

g. Any other area of facility operation considered appropriate by the CNRB or the Executive Vice President.

AUTHORITY

6.5.2.9 The CNRB shall report to and advise the Executive Vice President on those areas of responsibility specified in Sections 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 Section 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 Section 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.

6.6 REPORTABLE OCCURRENCE ACTION

6.6.1 The following actions shall be taken for REPORTABLE OCCURRENCES:

a. The Commission shall be notified and/or a report submitted pursuant to the requirements of Specification 6.9.

b. Each REPORTABLE OCCURRENCE requiring 24 hour notification to the Commission shall be reviewed by the FRG and submitted to the CNRB and the Vice President of Power Resources.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The facility shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President of Power Resources and to the CNRB within 24 hours.

c. 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.

d. The Safety Limit Violation Report shall be submitted to the Commission, the CNRB and the Director of Power Resources within 10 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

6.8.2 Each procedure and administrative policy of 6.8.1 above, and changes thereto, shall be reviewed by the FRG and approved by the Plant Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

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 7 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCES

6.9.1 Information to be reported to the Commission, in addition to the reports required by Title 10, Code of Federal Regulations, shall be in accordance with the Regulatory Position in Revision 4 of Regulatory Guide 1.16, "Reporting of Operating Information - Appendix "A" Technical Specifications."

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Director of the Office of Inspection and Enforcement Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification:

a. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

b. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

c. Inservice Inspection Program Reviews, Specifications 4.4.10.1 and 4.4.10.2.

d. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

e. Sealed Source leakage in excess of limits, Specification 4.7.9.1.3.

f. Seismic event analysis, Specification 4.3.3.3.2.

g. Beach survey results, Specification 4.7.6.1.1

h. Core Barrel Movement, Specifications 3.4.11 and 4.4.11.

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of facility 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 reactor tests and experiments.

f. Records of changes made to Operating Procedures.

g. Records of radioactive shipments.

h. Records of sealed source leak tests and results.

i. 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 Facility Operating License:

a. Records and drawing changes reflecting facility 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 facility radiation and contamination surveys.

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 facility components identified in Table 5.9-1

g. Records of training and qualification for current members of the plant staff.

h. Records of in-service inspections performed pursuant to these Technical Specifications.

i. Records of Quality Assurance activities required by the QA Manual.

j. Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k. Records of meetings of the FRG and the CNRB.

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 RESPIRATORY PROTECTION PROGRAM

ALLOWANCE

6.12.1 Pursuant to 10 CFR 20.103(c)(1) and (3), allowance may be made for the use of respiratory protective equipment in conjunction with activities authorized by the operating license for this facility in determining whether individuals in restricted areas are exposed to concentrations in excess of the limits specified in Appendix B, Table I, Column 1, of 10 CFR 20, subject to the following conditions and limitations:

a. The limits provided in Section 20.103(a) and (b) shall not be exceeded.

b. If the radioactive material is of such form that intake through the skin or other additional route is likely, individual exposures to radioactive material shall be controlled so that the radioactive content of any critical organ from all routes of intake averaged over 7 consecutive days does not exceed that which would result from inhaling such radioactive material for 40 hours at the pertinent concentration values provided in Appendix B, Table I, Column I, of 10 CFR 20.

c. For radioactive materials designated "Sub" in the "Isotope" column of Appendix B, Table I, Column 1 of 10 CFR 20, the concentration value specified shall be based upon exposure to the material as an external radiation source. Individual exposures to these materials shall be accounted for as part of the limitation on individual dose in §20.101. These materials shall be subject to applicable process and other engineering controls.

PROTECTION PROGRAM

6.12.2 In all operations in which adequate limitation of the inhalation of radioactive material by the use of process or other engineering controls is impracticable, the licensee may permit an individual in a restricted area to use respiratory protective equipment to limit the inhalation of airborne radioactive material, provided:

a. The limits specified in 6.12.1 above, are not exceeded.

b. Respiratory protective equipment is selected and used so that the peak concentrations of airborne radioactive material inhaled by an individual wearing the equipment do not exceed the pertinent concentration values specified in Appendix B, Table I, Column 1, of 10 CFR 20. For the purposes of this subparagraph, the concentration of radioactive material that is inhaled when respirators are worn may be determined by dividing the ambient airborne concentration by the protection factor specified in Table 6.12-1 for the respirator protective equipment worn. If the intake of radioactivity is later determined by other measurements to have been different than that initially estimated, the later quantity shall be used in evaluating the exposures.

c. The licensee advises each respirator user that he may leave the area at any time for relief from respirator use in case of equipment malfunction, physical or psychological discomfort, or any other condition that might cause reduction in the protection afforded the wearer.

d. The licensee maintains a respiratory protective program adequate to assure that the requirements above are met and incorporates practices for respiratory protection consistent with those recommended by the American National Standards Institute (ANSI-Z88.2-1969). Such a program shall include:

1. Air sampling and other surveys sufficient to identify the hazard, to evaluate individual exposures, and to permit proper selection of respiratory protective equipment.

2. Written procedures to assure proper selection, supervision, and training of personnel using such protective equipment.

3. Written procedures to assure the adequate fitting of respirators; and the testing of respiratory protective equipment for OPERABILITY immediately prior to use.

4. Written procedures for maintenance to assure full effectiveness of respiratory protective equipment, including issuance, cleaning and decontamination, inspection, repair, and storage.

5. Written operational and administrative procedures for proper use of respiratory protective equipment including provisions for planned limitations on working times as necessitated by operational conditions.

6. Bioassays and/or whole body counts of individuals (and other surveys, as appropriate) to evaluate individual exposures and to assess protection actually provided.

e. The licensee uses equipment approved by the U.S. Bureau of Mines under its appropriate Approval Schedules as set forth in Table 6.12-1. Equipment not approved under U.S. Bureau of Mines Approval Schedules shall be used only if the licensee has evaluated the equipment and can demonstrate by testing, or on the basis of reliable test information, that the material and performance characteristics of the equipment are at least equal to those afforded by U.S. Bureau of Mines approved equipment of the same type, as specified in Table 6.12-1.

f. Unless otherwise authorized by the Commission, the licensee shall not assign protection factors in excess of those specified in Table 6.12-1 in selecting and using respiratory protective equipment.

REVOCATION

6.12.3 The specifications of Section 6.12 shall be revoked in their entirety upon adoption of the proposed change to 10 CFR 20, Section 20.103, which would make such provisions unnecessary.

6.13 HIGH RADIATION AREA

6.13.1 In lieu of the "control device" or "alarm signal" required by paragraph 20.203(c)(2) of 10 CFR 20:

a. A 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 issuance of a Radiation Work Permit and any individual or group of individuals permitted to enter such areas shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

b. A High Radiation Area in which the intensity of radiation is greater than 1000 mrem/hr shall be subject to the provisions of 6.13.1.a above, and in addition locked doors shall be provided to prevent unauthorized entry into such areas and the keys shall be maintained under the administrative control of the Shift Supervisor on duty.

TABLE 6.12-1

PROTECTION FACTORS FOR RESPIRATORS

DESCRIPTION	MODES[1]	PROTECTION FACTORS[2] PARTICULATES AND VAPORS AND GASES EXCEPT TRITIUM OXIDE[3]	GUIDES TO SELECTION OF EQUIPMENT BUREAU OF MINES/NATIONAL INSTITUTE FOR OCCUPATIONAL SAFETY AND HEALTH APPROVALS
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask[4,7]	NP	5	30 CFR Part 11 Subpart K
Facepiece, full[7]	NP	100	30 CFR Part 11 Subpart K
II. ATMOSPHERE-SUPPLYING RESPIRATOR			
1. Airline respirator			
Facepiece, half-mask	CF	100	30 CFR Part 11 Subpart J
Facepiece, full	CF	1,000	30 CFR Part 11 Subpart J
Facepiece, full[7]	D	100	30 CFR Part 11 Subpart J
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart J
Hood	CF	[5]	30 CFR Part 11 Subpart J
Suit	CF	[5]	[6]
2. Self-contained breathing apparatus (SCBA)			
Facepiece, full[7]	D	100	30 CFR Part 11 Subpart H
Facepiece, full	PD	1,000	30 CFR Part 11 Subpart H
Facepiece, full	R	100	30 CFR Part 11 Subpart H
III. COMBINATION RESPIRATOR Any combination of air-purifying and atmosphere-supplying respirator		Protection factor for type and mode of operation as listed above	30 CFR Part 11 § 11.63(b)

TABLE 6.12-1 (Continued)

TABLE NOTATION

[1] See the following symbols:

- CF: continuous flow
- D: demand
- NP: negative pressure (i.e., negative phase during inhalation)
- PD: pressure demand (i.e., always positive pressure)
- R: recirculating (closed circuit)

[2](a) For purposes of this specification the protection factor is a measure of the degree of protection afforded by a respirator, defined as the ratio of the concentration of airborne radioactive material outside the respiratory protective equipment to that inside the equipment (usually inside the facepiece) under conditions of use. It is applied to the ambient airborne concentration to estimate the concentration inhaled by the wearer according to the following formula:

$$\text{Concentration Inhaled} = \frac{\text{Ambient Airborne Concentration}}{\text{Protection Factor}}$$

(b) The protection factors apply:

(i) only for trained individuals wearing properly fitted respirators used and maintained under supervision in a well-planned respiratory protective program.

(ii) for air-purifying respirators only when high efficiency [above 99.9% removal efficiency by U.S. Bureau of Mines type dioctyl phthalate (DOP) test] particulate filters and/or sorbents appropriate to the hazard are used in atmospheres not deficient in oxygen.

(iii) for atmosphere-supplying respirators only when supplied with adequate respirable air.

[3] Excluding radioactive contaminants that present an absorption or submersion hazard. For tritium oxide approximately half of the intake occurs by absorption through the skin so that an overall protection factor of not more than approximately 2 is appropriate when atmosphere-supplying respirators are used to protect against tritium oxide. Air-purifying respirators are not recommended for use against tritium oxide. See also footnote [5], below, concerning supplied-air suits and hoods.

TABLE 6.12-1 (Continued)

TABLE NOTATION

[4] Under chin type only. Not recommended for use where it might be possible for the ambient airborne concentration to reach instantaneous values greater than 50 times the pertinent values in Appendix B, Table I, Column 1 of 10 CFR Part 20.

[5] Appropriate protection factors must be determined taking account of the design of the suit or hood and its permeability to the contaminant under conditions of use. No protection factor greater than 1,000 shall be used except as authorized by the Commission.

[6] No approval schedules current available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

[7] Only for shaven faces.

NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines or the National Institute for Occupational Safety and Health according to approval schedules for respirators to protect against airborne radionuclides, may be used to the extent that they do not exceed the protection factors listed in this Table. The protection factors in this Table may not be appropriate to circumstances where chemical or other respiratory hazards exist in addition to radioactive hazards. The selection and use of respirators for such circumstances should take into account approvals of the U.S. Bureau of Mines or the National Institute for Occupational Safety and Health in accordance with its applicable schedules.

NOTE 2: Radioactive contaminants for which the concentration values in Appendix B, Table I of 10 CFR Part 20 are based on internal dose due to inhalation may, in addition, present external exposure hazards at higher concentrations. Under such circumstances, limitations on occupancy may have to be governed by external dose limits.

APPENDIX B

TO

OPERATING LICENSE NO. DPR-67

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

FOR

Florida Power & Light Company

St. Lucie Unit No. 1

Docket No. 50-335

MAR 01 1976

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1.0 DEFINITIONS

The definitions for terms used in these environmental technical specifications are listed below.

1.1 National Power Emergency

Shall mean any event causing authorized Federal officials to require or request that Florida Power and Light supply electricity to points within or without the State of Florida.

1.2 A Regional Emergency

Shall mean any of the following occurrences within the State of Florida: (1) a catastrophic natural disaster including hurricanes, floods, and tidal waves; or (2) other emergencies declared by State, county, municipal, or Federal authorities during which an uninterrupted supply of electric power is vital to public health and safety.

1.3 Reactor Emergency

Shall mean an unanticipated equipment malfunction necessitating prompt remedial action to avoid endangering the public health or safety.

1.4 Circulating Water System

Comprised of the following; velocity cap, intake pipe, intake canal, discharge canal, discharge pipe, "Y" port discharge and miscellaneous mechanical devices. The recirculation canal is included, if constructed.

1.5 Frequency Definitions

Daily - Not less than 360 times per annum.

Weekly - Not less than 48 times per annum - interval may vary by 3 days.

Monthly - Not less than 12 times per annum - interval may vary by 15 days.

Quarterly - Not less than 4 times per annum - interval may vary by 30 days.

Semi-annually - Not less than 2 times per annum - interval may vary by 60 days.

Refueling - at refueling intervals not to exceed 24 months.

1.6 Total Residual Chlorine

The amount of free and combined available chlorine present in water.

1.7 Intake Temperature

The temperature of the cooling water as measured at the plant intake structure.

1.8 Discharge Temperature

The temperature of the cooling water as measured near the terminus of the discharge canal.

1.9 Dissolved Oxygen

Oxygen dissolved in the condenser cooling water, and expressed in milligrams per liter.

1.10 Limiting Conditions

Those conditions to be imposed on plant effluents and operating practices which may have an adverse impact on the environment.

1.11 Salinity

The total amount of solid material in grams contained in one kilogram of sea water when all the carbonate has been converted to oxide, the bromine and iodine replaced by chlorine, and organic matter completely oxidized.

1.12 Continuous Recording

Recording of a measured parameter on a chart by a single pen or a multi-point recorder with less than a one-minute interval between successive printing of the same parameter.

1.13 Channel Calibration

A Channel Calibration shall be the adjustment of the channel output such that it corresponds with specified 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.

1.14 Channel Functional Test

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.

1.15 Batch Releases

Discontinuous release of gaseous or liquid effluent which takes place over a finite period of time, usually hours or days.

1.16 Continuous Release

Release of gaseous or liquid effluent which is essentially uninterrupted for extended periods during normal operation of the facility.

2.0 LIMITING CONDITIONS

General

2.0.1 The circulating water system shall be operated to result in an acceptable environmental impact. Flexibility of operation is permitted, consistent with consideration of health and safety, to ensure that the public is provided a dependable source of power even under unusual operating conditions which may set forth in this specification, as provided below in 2.0.2 and 2.0.3.

2.0.2 During a national power emergency, a regional emergency, reactor emergency, or any time when the health or safety of the public may be endangered by the inability of Florida Power and Light to supply electricity from any other sources available to it, the operating limits provided in this specification shall be inapplicable. However, during such emergencies, the operating limits shall not be exceeded except as is necessitated by the emergency.

2.0.3 Whenever, in accordance with paragraphs 2.0.1 and 2.0.2 above, Florida Power and Light exceeds the operating limits otherwise imposed, notification shall be made to the Director of the Region II Regional Office of the Office of Inspection and Enforcement, in accordance with 5.6.2.a.

2.1 THERMAL

2.1.1 Maximum Discharge Temperature

Objective

The purpose of this specification is to limit thermal stress to the aquatic ecosystem by limiting the temperature rise in the Atlantic Ocean, in the area of the subaqueous discharge, during operation.

Specification

The thermal discharge of St. Lucie Unit No. 1 into the Atlantic Ocean shall be limited to a maximum release temperature of 111°F and shall not cause a temperature rise in excess of 1.5°F above ambient surface temperature outside a 400 acre zone of mixing during the months of June through September, nor a 4°F rise during the remaining months. In addition, the surface temperature conditions within the zone of mixing shall not exceed a rise of 5.5°F over ambient temperature nor a maximum temperature of 93°F as an instantaneous maximum at any point.

Thermal defouling of the intake pipeline is allowed subject to a maximum release temperature of 120°F and a maximum surface temperature rise of 2°F.

Under the following conditions, which may be expected to cause the discharge temperature to be higher than design, the maximum discharge temperature shall be limited to 115°F: 1) Condenser and/or circulating water pump maintenance; 2) Throttling circulating water pumps to minimize use of chlorine; 3) Fouling of circulating water system.

Temporary transients due to accidental loss of circulating water system components may cause temperature rises in excess of limitations stated above. Variances due to these transients shall be limited to no more than 7 hours per month.

Monitoring Requirement

A continuous temperature measurement system shall be installed in the discharge canal at approximately mid-depth. Temperatures shall be transmitted to the control room.

A continuous temperature monitoring station located within 500 feet from the primary monitoring device shall be used as a backup system if the primary system fails. In this event this station shall be checked every 8 hours until the primary system is restored. See Section 3.1.A.6 for complete details of the monitoring program.

2.1.2 Maximum Condenser Temperature Rise

Under normal full power operation, the temperature rise across the condenser shall not exceed 24°F. Under the following conditions, the condenser temperature rise shall not exceed 35°F for greater than a 72-hour period: 1) Condenser and/or circulating water pump maintenance; 2) Throttling circulating water pumps to minimize use of chlorine; 3) Fouling of circulating water system.

Monitoring Requirements

The ΔT across the condenser shall be determined once per hour while the unit is in operation. The system's accuracy and precision is as described in Section 3.1.A.6 of Appendix B of the technical specifications.

Bases

The limitations provide reasonable assurance that the overall aquatic ecosystem in the area of the thermal plume will experience an acceptable environmental impact. The placement of the temperature monitoring instrument in the discharge canal will give the temperature of the discharge water before mixing with the receiving water.

2.2 CHEMICAL

Objective

The purpose of these specifications is 1) to minimize impacts to the quality of the Atlantic Ocean, 2) to protect the local biota from lethal and sublethal effects of exposure to chemical discharges due to operation of the plant, 3) to assure that continued multiple use of the receiving waters by human populations is protected, and 4) to control the quality of the receiving medium.

2.2.1 Biocides

Specification

Total Residual Chlorine shall not exceed 0.1 mg/l at any time at the terminus of the discharge canal (prior to entering the ocean outfall). If this level is exceeded, adjustments to the injection system shall be made to reduce the concentration, and each succeeding chlorination period shall be monitored until the concentration is within the specification. Chlorine shall not be added for more than 2 hours per day.

Monitoring Requirements

A grab sample of condenser cooling water shall be taken weekly in the discharge canal and analyzed for total residual chlorine. The samples shall be taken during the period of chlorination. The time of beginning the chlorination and when the sample was taken shall be logged.

Bases

When injected, chlorine is diluted by the cooling water and consumed in the process of controlling slime. To be sure that enough chlorine is injected to control the slime, the residual chlorine concentration will be approximately 1 mg/l at the condenser outlet.

2.2.2 pH

Specification

The pH of the cooling water in the discharge canal shall not be less than 6.0 nor greater than 9.0 pH units.

Monitoring Requirement

pH shall be measured on a daily basis in the discharge canal, and it shall be accomplished using either a grab sample or recorder.

Bases

The pH limits set forth will provide reasonable assurance of an acceptable environmental impact when discharging waters to the Atlantic Ocean.

2.3 RESERVED

2.4 RADIOACTIVE EFFLUENTS

Objective

To define the limits and conditions for the controlled release of radioactive materials in liquid and gaseous effluents to the environs to ensure that

these releases are as low as practicable. These releases should not result in radiation exposures in unrestricted areas greater than a few percent of natural background exposures. The concentration of radioactive materials in effluents shall be within the limits specified in 10 CFR Part 20.

To ensure that the releases of radioactive material above background to unrestricted areas be as low as practicable, the following design objectives apply:

For liquid wastes:

a. The annual dose above background to the total body or any organ of an individual from all reactors at a site should not exceed 5 mrem in an unrestricted area.

b. The annual total quantity of radioactive materials in liquid waste, excluding tritium and dissolved gases, discharged from each reactor should not exceed 5 Ci.

For gaseous wastes:

c. The annual total quantity of noble gases above background discharged from the site should result in an annual air dose due to gamma radiation of less than 10 mrad, and an annual air dose due to beta radiation of less than 20 mrad, at any location near ground level which could be occupied by individuals at or beyond the boundary of the site, and that no individual in an unrestricted area will receive an annual dose to the total body greater than 5 mrem or an annual skin dose greater than 15 mrem from this release quantity.

d. The annual total quantity of all radioiodines and radioactive material in particulate forms with half-lives greater than eight days, above background, from all reactors at a site should not result in an annual dose to any organ of an individual in an unrestricted area from all pathways of exposure in excess of 15 mrem.

e. The annual total quantity of iodine-131 discharged from each reactor at a site should not exceed 1 Ci.

2.4.1 Liquid Waste Effluents

a. The concentration of radioactive materials released in liquid waste effluents from all reactors at the site shall not exceed the value specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for unrestricted areas.

b. The cumulative release of radioactive materials in liquid waste effluents, excluding tritium and dissolved gases, shall not exceed 10 Ci/reactor/calendar quarter.

c. The cumulative release of radioactive materials in liquid waste effluents, excluding tritium and dissolved gases, shall not exceed 20 Ci/reactor in any 12 consecutive months.

d. During release of radioactive wastes, the effluent control monitor shall be set to alarm and to initiate the automatic closure of each waste isolation valve prior to exceeding the limits specified in 2.4.1.a above, except as provided in 2.4.2.d below.

e. The operability of each automatic isolation valve in the liquid radwaste discharge lines shall be demonstrated quarterly.

f. The equipment installed in the liquid radioactive waste system shall be maintained and shall be operated to process radioactive liquid wastes prior to their discharge when the projected cumulative release could exceed 1.25 Ci/reactor/calendar quarter, excluding tritium and dissolved gases.

g. The maximum radioactivity to be contained in any liquid radwaste tank that can be discharged directly to the environs shall not exceed 10 Ci, excluding tritium and dissolved gases.

h. If the cumulative release of radioactive materials in liquid effluents, excluding tritium and dissolved gases, exceeds 2.5 Ci/reactor/calendar quarter, the licensee shall make an investigation to identify the causes for such releases, define and initiate a program of action to reduce such releases to the design objective levels listed in Section 2.4, and report these actions to the NRC in accordance with Specification 5.6.2.b(1).

i. An unplanned or uncontrolled offsite release of radioactive materials in liquid effluents in excess of 0.5 curies requires notification. This notification shall be in accordance with Specification 5.6.2.b(3).

2.4.2 <u>Liquid Waste Sampling and Monitoring</u>

a. Plant records shall be maintained of the radioactive concentration and volume before dilution of liquid waste intended for discharge and the average dilution flow and length of time over which each discharge occurred. Sample analysis results and other reports shall be submitted as required by Section 5.6.1 of these Specifications. Estimates of the sampling and analytical errors associated with each reported value shall be included.

b. Prior to release of each batch of liquid waste, a sample shall be taken from that batch and analyzed for the concentration of each significant gamma energy peak in accordance with Table 2.4-1 to demonstrate compliance with Specification 2.4.1 using the flow rate into which the waste is discharged during the period of discharge.

c. Sampling and analysis of liquid radioactive waste shall be performed in accordance with Table 2.4-1.

TABLE 2.4-1
RADIOACTIVE LIQUID SAMPLING AND ANALYSIS

Liquid Source	Sampling Frequency	Type of Activity Analysis	Detectable Concentrations (μCi/ml)[1]
A. Monitor Tank Releases	Each Batch	Principal Gamma Emitters[6]	5×10^{-7}[2]
	One batch/month	Dissolved Gases[5]	10^{-5}
	Weekly Composite[3]	Ba-La-140, I-131	10^{-6}
	Monthly Composite[3]		
		H-3	10^{-5}
		Gross α	10^{-7}
	Quarterly Composite[3]	Sr-90, Sr-89	5×10^{-3}
B. Primary Coolant	As required in Appendix A of the technical specifications[8]	I-131, I-133	10^{-6}
C. Steam Generator Blowdown Releases	weekly[4,6,7]	Principal Gamma emitters	5×10^{-7}[2]
		Ba-La-140, I-131	10^{-6}
	One Sample/ Month	Dissolved Gases[5]	10^{-5}
	Monthly Composite[4]		
		H3	10^{-5}
		Gross α	10^{-7}
	Quarterly Composite[4]	Sr-90, Sr-89	5×10^{-3}
D. Service Water[9] Discharge Pipe	weekly[4,6,7]	Principal Gamma entitters	5×10^{-7}[2]
		Ba-La-140, I-131	10^{-6}
	One Sample/ Month	Dissolved Gases[5]	10^{-5}
	Monthly Composite[4]	H-3	10^{-5}
		Gross α	10^{-7}
	Quarterly Composite[4]	Sr-90, Sr-89	5×10^{-3}

TABLE 2.4-1 (Cont'd)

[1]The detectability limits for activity analysis are based on the technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable, and when nuclides are measured below the stated limits, they should also be reported.

[2]For certain mixtures of gamma emitters, it may not be possible to measure radionuclides in concentrations near their sensitivity limits when other nuclides are present in the sample in much greater concentrations: Under these circumstances, it will be more appropriate to calculate the concentrations of such radionuclides using measured ratios with those radionuclides which are routinely identified and measured.

[3]A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged.

[4]To be representative of the average quantities and concentrations of radioactive materials in liquid effluents, samples should be collected in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite should be thoroughly mixed in order for the composite sample to be representative of the average effluent release.

[5]For dissolved noble gases in water, assume a MPC of 4×10^{-5} μCi/ml of water.

[6]When operational or other type of limitations preclude specific gamma spectrum analysis of each tank, gross activity measurements shall be made to estimate the quantity and concentration of radioactive material released in the batch and a weekly sample composited from proportional aliquots from each batch released during the week shall be analyzed for the principal gamma emitting radionuclides.

[7]No sampling required when cold and drained.

[8]Should the reactor coolant system activity technical specification limits be exceeded, the power level and cleanup or purification flow rate at the sample time shall also be reported.

[9]Required if the component cooling water (CCW) monitor is out of service or if the CCW monitor indicates activity in excess of 10^{-5} μCi/cc.

d. The radioactivity in liquid wastes shall be continuously monitored and recorded during release. Whenever these monitors are inoperable for a period not to exceed 72 hours, two independent samples of each tank to be discharged shall be analyzed and two plant personnel shall independently check valving prior to the discharge. If these monitors are inoperable for a period exceeding 72 hours, no release from a liquid waste tank shall be made and any release in progress shall be terminated.

e. The flow rate of liquid radioactive waste shall be continuously measured and recorded during release. If the flow monitors are inoperable, the release flow shall be determined and recorded each hour based upon an estimate of the flow rate of the system.

f. All liquid effluent radiation monitors shall be calibrated at least quarterly by means of a radioactive source which has been calibrated to a National Bureau of Standards source. Each monitor shall also have a functional test monthly and an instrument check prior to making a release.

g. The radioactivity in steam generator blowdown shall be continuously monitored and recorded. With one steam generator blowdown monitor inoperable, the sampling system shall be realigned so that the operable monitor is receiving flow from both steam generators. Whenever both monitors are inoperable, the blowdown flow shall be diverted to the waste management system and the direct release to the environment terminated.

h. The points of release to the environment to be monitored in this section 2.4.2 include all the monitored release points as provided for in Table 2.4-2.

Bases

The release of radioactive materials in liquid waste effluents to unrestricted areas shall not exceed the concentration limits specified in 10 CFR Part 20 and should be as low as practicable in accordance with the requirements of 10 CFR Part 50.36a. These specifications provide reasonable assurance that the resulting annual dose to the total body or any organ of an individual in an unrestricted area will not exceed 5 mrem. At the same time, these specifications permit the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. It is expected that by using this operational flexibility under unusual operating conditions, and exerting every effort to keep levels of radioactive material in liquid wastes as low as practicable, the annual releases will not exceed a small fraction of the concentration limits specified in 10 CFR Part 20.

TABLE 2.4-2

ST. LUCIE PLANT LIQUID WASTE SYSTEM
LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLERS REQUIRED BY TECHNICAL SPECIFICATIONS

Process Stream or Release Point	Radiation Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	Measurement						High Liquid Level Alarm
					Gross Activity	I	Dissolved Gases	Alpha	H-3	Isotopic Analysis	
Miscellaneous Waste Sample (Test) Tank (Waste and Boric Acid Condensate Tanks)				X		X	X	X	X	X	X
Detergent Waste Collector Tank (Laundry Drain Tanks)				X		X	X	X	X	X	X
Primary Coolant System				X		X					
Liquid Radwaste Discharge Pipe	X	X	X		X						
Steam Generator Blow-down System	X		X	X	X	X	X	X	X	X	
Outdoor Storage Tanks (potentially radio-active)				X	X					X	X
Component Cooling Systems	X		X		X						
Turbine Building Floor Drains (Storm Drains)				X[a]	X[a]						

[a] Grab samples to be taken and analyzed each 8 hours when the gross activity in the secondary coolant system exceeds 10^{-5} μCi/ml.

The design objectives have been developed based on operating experience taking into account a combination of variables including defective fuel, primary system leakage, primary to secondary system leakage, steam generator blowdown and the performance of the various waste treatment systems, and are consistent with 10 CFR Part 50.36a.

Specification 2.4.1.a requires the licensee to limit the concentration of radioactive materials in liquid waste effluents released from the site to levels specified in 10 CFR Part 20, Appendix B, Table II, Column 2, for unrestricted areas. This specification provides assurance that no member of the general public will be exposed to liquid containing radioactive materials in excess of limits considered permissible under the Commission's Regulations.

Specifications 2.4.1.b and 2.4.1.c establish the upper limits for the release of radioactive materials in liquid effluents. The intent of these specifications is to permit the licensee the flexibility of operation to assure that the public is provided a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the levels normally achievable when the plant and the liquid waste treatment systems are functioning as designed. Releases of up to these levels will result in concentrations of radioactive material in liquid waste effluents at small percentages of the limits specified in 10 CFR Part 20.

Consistent with the requirements of 10 CFR Part 50, Appendix A, Design Criterion 64, Specifications 2.4.1.d and 2.4.1.e require operation of suitable equipment to control and monitor the releases of radioactive materials in liquid wastes during any period that these releases are taking place.

Specification 2.4.1.f requires that the licensee maintain and operate the equipment installed in the liquid waste systems to reduce the release of radioactive materials in liquid effluents to as low as practicable consistent with the requirements of 10 CFR Part 50.36a. Normal use and maintenance of installed equipment in the liquid waste system provides reasonable assurance that the quantity released will not exceed the design objective. In order to keep releases of radioactive materials as low as practicable, the specification requires operation of equipment whenever it appears that the projected cumulative discharge rate will exceed one-fourth of this design objective annual quantity during any calendar quarter.

Specification 2.4.1.g restricts the amount of radioactive material that could be inadvertently released to the environment to an amount that will not exceed the Technical Specification limit.

In addition to limiting conditions for operation listed under Specifications 2.4.1.b and 2.4.1c, the reporting requirements of Specification 2.4.1.h delineate that the licensee shall identify the cause whenever the cumulative release of radioactive materials in liquid waste effluents

exceeds one-half the design objective annual quantity during any calendar quarter and describe the proposed program of action to reduce such releases to design objective levels on a timely basis. This report must be filed within 30 days following the calendar quarter in which the release occurred as required by Specification 5.6.2 of these Technical Specifications.

Specification 2.4.1.i provides for reporting spillage or release events which, while below the limits of 10 CFR Part 20, could result in releases higher than the design objectives.

The sampling and monitoring requirements given under Specification 2.4.2 provide assurance that radioactive materials in liquid wastes are properly controlled and monitored in conformance with the requirements of Design Criteria 60 and 64. These requirements provide the data for the licensee and the Commission to evaluate the plant's performance relative to radioactive liquid wastes released to the environment. Reports on the radioactive materials released in liquid waste effluents are furnished to the Commission according to Section 5.6.1 of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

The points of release to the environment to be monitored in Section 2.4.2 include all the monitored release points as provided for in Table 2.4-2.

2.4.3 Gaseous Waste Effluents

The terms used in these Specifications are as follows:

subscript v, refers to vent releases

subscript i, refers to individual noble gas nuclide (Refer to Table 2.4-3 for the noble gas nuclides considered)

Q_T = the total noble gas release rate (Ci/sec)

= $\sum_i Qi$ sum of the individual noble gas radionuclides determined to be present by isotopic analysis

$\overline{K}$ = the average total body dose factor due to gamma emission (rem/yr per Ci/sec)

$\overline{L}$ = the average skin dose factor due to beta emissions (rem/yr per Ci/sec)

$\overline{M}$ = the average air dose factor due to beta emissions (rad/yr per Ci/sec)

$\overline{N}$ = the average air dose factor due to gamma emissions (rad/yr per Ci/sec)

TABLE 2.4-3

GAMMA AND BETA DOSE FACTORS FOR

ST. LUCIE PLANT, UNIT 1

$X/Q = 2.1 \times 10^{-6}$ sec/m^3

NOBLE GAS RADIONUCLIDE	DOSE FACTORS FOR VENT			
	K_{iv} Total Body $\frac{rem/yr}{Ci/sec}$	L_{iv} Skin $\frac{rem/yr}{Ci/sec}$	M_{iv} Beta Air $\frac{rad/yr}{Ci/sec}$	N_{iv} Gamma Air $\frac{rad/yr}{Ci/sec}$
Kr-83m	5.8×10^{-5}	0	0.6	0.028
Kr-85m	0.88	3.1	4.1	0.92
Kr-85	0.014	2.8	4.1	0.015
Kr-87	1.9	20	22	2.0
Kr-88	6.0	5.0	6.2	6.3
Kr-89	0.5	21	22	0.52
Xe-131m	0.4	1.0	2.3	0.5
Xe-133m	0.3	2.1	3.1	0.41
Xe-133	0.36	0.64	2.2	0.45
Xe-135m	0.64	1.5	1.6	0.68
Xe-135	1.5	3.9	5.2	1.6
Xe-137	0.072	26	27	0.076
Xe-138	1.5	8.7	10	1.6

The values of $\overline{K}$, $\overline{L}$, $\overline{M}$ and $\overline{N}$ are to be determined each time isotopic analysis is required as delineated in Specification 2.4.4. Determine the following using the results of the noble gas radionuclide analysis:

$$\overline{K} = (1/Q_T)\sum_i Q_i K_i$$

$$\overline{L} = (1/Q_T)\sum_i Q_i L_i$$

$$\overline{M} = (1/Q_T)\sum_i Q_i L_i$$

$$\overline{N} = (1/Q_T)\sum_i Q_i N_i$$

where the values of K_i, L_i, M_i and N_i are provided in Table 2.4-3, and are site dependent gamma and beta dose factors

Q = the measured release rate of the radioiodines and radioactive materials in particulate forms with half-lives greater than eight days.

a. (1) The release rate limit of noble gases from the site shall be such that

$$2.0\left(Q_{Tv}\overline{K}_v\right) \leq 1$$

and

$$0.33\left(Q_{Tv}(\overline{L}_v + 1.1\overline{N}_v)\right) \leq 1$$

(2) The release rate limit of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days, released to the environs as part of the gaseous wastes from the site shall be such that

$$5.5 \times 10^3\ Q_v \leq 1$$

b. (1) The average release rate of noble gases from the site during any calendar quarter shall be such that

$$13\left(Q_{Tv}\overline{N}_v\right) \leq 1$$

and

$$6.3\left(Q_{Tv}\overline{M}_v\right) \leq 1$$

(2) The average release rate of gases from the site during any 12 consecutive months shall be

$$25 \left(Q_{Tv}\overline{N}_v \right) \leq 1$$

and

$$13 \left(Q_{Tv}\overline{M}_v \right) \leq 1$$

(3) The average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter shall be such that

$$13 \left(5.5x10^3 Q_v \right) \leq 1$$

(4) The average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any period of 12 consecutive months shall be such that

$$25 \left(5.5x10^3 Q_v \right) \leq 1$$

(5) The amount of iodine-131 released during any calendar quarter shall not exceed 2 Ci/reactor.

(6) The amount of iodine-131 released during any period of the 12 consecutive months shall not exceed 4 Ci/reactor.

c. Should any of the conditions of 2.4.3.c(1), (2) or (3) listed below exist, the licensee shall make an investigation to identify the causes of the release rates, define and initiate a program of action to reduce the release rates to design objective levels listed in Section 2.4 and report these actions to the NRC within 30 days from the end of the quarter during which the releases occurred.

(1) If the average release rate of noble gases from the site during any calendar quarter is such that

$$50 \left(Q_{Tv}\overline{N}_v \right) > 1$$

or

$$25 \left(Q_{Tv}\overline{M}_v \right) > 1$$

(2) If the average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter is such that

$$50 \left(5.5x10^3 Q_v \right) > 1$$

(3) If the amount of iodine-131 released during any calendar quarter is greater than 0.5 Ci/reactor.

d. During the release of gaseous wastes from the gas decay tanks, the gaseous discharge monitor shall be operating and set to alarm and to initiate the automatic closure of the waste gas discharge valve prior to exceeding the limits specified in 2.4.3.a above. Whenever this monitor is inoperable for a period not to exceed seven days, two independent samples of each gas decay tank to be discharged shall be analyzed and two plant personnel shall independently check valving prior to the discharge. If this monitor is inoperable for a period exceeding seven days, no release from a gas decay tank shall be made and any release in progress shall be terminated. The operability of each automatic isolation valve shall be demonstrated quarterly.

e. The maximum activity to be contained in one waste gas storage tank shall not exceed 110,000 curies (considered as Xe-133).

f. An unplanned or uncontrolled offsite release of radioactive materials in gaseous effluents in excess of 5 curies of noble gas or 0.02 curie of radioiodine in gaseous form requires notification. This notification shall be in accordance with Specification 5.6.2.b(3).

2.4.4 Gaseous Waste Sampling and Monitoring

a. Plant records shall be maintained and reports of the sampling and analyses results shall be submitted in accordance with Section 5.6 of these Specifications. Estimates of the sampling and analytical error associated with each reported value should be included.

b. Gaseous releases to the environment, except from the turbine building ventilation exhaust and as noted in Specification 2.4.4.c, shall be continuously monitored for gross radioactivity. Whenever these monitors are inoperable, grab samples shall be taken and analyzed daily for gross radioactivity. If these monitors are inoperable for more than seven days, these releases shall be terminated.

c. During the release of gaseous wastes from the primary system waste gas holdup system, the iodine collection device, and the particulate collection device shall be operating, except as noted in 2.4.3.d., above.

d. All waste gas effluent monitors shall be calibrated at least quarterly by means of a known radioactive source which has been calibrated to a National Bureau of Standards source. Each monitor shall have a functional test at least monthly and instrument check at least daily.

e. Sampling and analysis of radioactive material in gaseous waste, including particulate forms and radioiodines shall be performed in accordance with Table 2.4-2. The points of release to the environment to be monitored in this Section 2.4.4 include all the monitored release points as provided for in Table 2.4-4.

TABLE 2.4-4
RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS

Gaseous Source	Sampling Frequency	Type of Activity Analysis	Detectable Concentration (μCi/ml)[1]
A. Waste Gas Decay Tank Releases	Each Tank	Principal Gamma Emitters	10^{-4}[2]
		H-3	10^{-6}
B. Containment Purge Releases	Each Purge	Principal Gama Emitters	10^{-4}[3]
		H-3	10^{-6}
C. Condenser Air Ejector	Monthly[5]	Principal Gamma Emitters	10^{-4}[2,3]
		H-3	10^{-6}
D. Environmental Release Points	Monthly (Gas Samples)	Principal Gamma Emitters	10^{-4}[2,3]
		H-3	10^{-6}
	Weekly (Charcoal Sample)	I-131	10^{-12}
	Monthly (Charcoal Sample)	I-133, I-135	10^{-10}
	Weekly (Particulates)	Principal Gamma Emitters (Ba-La-140, I-131, and others)	10^{-11}
	Monthly Composite[4] (Particulates)		
		Gross α	10^{-11}
	Quarterly Composite[4] (Particulates)	Sr-90, Sr-89	10^{-11}

[1] The above detectability limits for activity and analysis are based on technical feasibility and on the potential significance in the environment of the quantities released. For some nuclides, lower detection limits may be readily achievable, and when nuclides are measured below the stated limits, they should also be reported.

[2] For certain mixtures of gamma emitters, it may not be possible to measure radionuclides at levels near their sensitivity limits when other nuclides are present in the sample at much higher levels. Under these circumstances, it will be more appropriate to calculate the levels of such radionuclides using observed ratios with those radionuclides which are measurable.

[3] Analyses shall also be performed following each refueling, startup, or similar operational occurrence which could alter the mixture of radionuclides.

[4] To be representative of the average quantities and concentrations of radioactive materials in particulate form released in gaseous effluents, samples should be collected in proportion to the rate of flow of the effluent stream.

[5] Required when the gross activity in the secondary coolant system, as required to be determined in Appendix A of these technical specifications, exceeds 10^{-5} uCi/ml.

TABLE 2.4-5

ST. LUCIE PLANT GASEOUS WASTE SYSTEM
LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLERS REQUIRED BY TECHNICAL SPECIFICATIONS

Process Stream or Release Point	Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	Measurement: Noble Gas	I	Particulate	H-3	Alpha
Waste Gas Storage Tanks	X	X	X	X	X	X	X	X	X
Condenser Air Ejector	X		X	X	X	X	X	X	X
Building Ventilation Systems									
Reactor Containment Building[a] (whenever there is flow to Plant Vent)	X		X	X	X	X	X	X	X
Auxiliary Building[a] (to Plant Vent)	X		X	X	X	X	X	X	X
Fuel Handling & Storage Building[a]	X		X	X	X	X	X	X	X
Radwaste Area[a] (to Plant Vent)	X		X	X	X	X	X	X	X
Steam Generator Blowdown Tank Vent or Condenser Vent[b]	X(d)		X(d)	X	X	X	X	X	X
Turbine Gland Seal Condenser[e]	X		X	X	X	X	X	X	X
Waste Evaporator Condenser Vent[c,e]	X		X	X	X	X	X	X	X

[a] If any or all of the process streams or building ventilation systems are routed to a single release point, the need for a continuous monitor at the individual discharge point to the main exhaust duct is eliminated. One continuous monitor at the final release point is sufficient.

[b] In some PWRs the steam generator blowdown tank vent is routed to the main turbine condenser, and the need for a continuous monitor at this release point is eliminated.

[c] For PWRs in which the waste evaporator condenser is vented directly to the atmosphere.

[d] Monitoring system will be installed by March 1, 1976, if this system is still operational.

[e] Monitored via Condenser Air Ejector System.

Bases

The release of radioactive materials in gaseous waste effluents to unrestricted areas shall not exceed the concentration limits specified in 10 CFR Part 20 and should be as low as practical in accordance with the requirements of 10 CFR Part 50.36a. These Specifications provide reasonable assurance that the resulting annual air dose from the site due to gamma radiation will not exceed 10 mrad, and an annual air dose from the site due to beta radiation will not exceed 20 mrad from noble gases, that no individual in an unrestricted area will receive an annual dose to the total body greater than 5 mrem or an annual skin dose greater than 15 mrem from fission product noble gases, and that the annual dose to any organ of an individual from radioiodines and radioactive material in particulate form with half-lives greater than eight days will not exceed 15 mrem per site.

At the same time these Specifications permit the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided with a dependable source of power under unusual operating conditions which may temporarily result in releases higher than the design objective levels but still within the concentration limits specified in 10 CFR Part 20. Even with this operational flexibility under unusual operating conditions, if the licensee exerts every effort to keep levels of radioactive material in gaseous waste effluents as low as practicable, the annual releases will not exceed a small fraction of the concentration limits specified in 10 CFR Part 20.

The design objectives have been developed based on operating experience taking into account a combination of system variables including defective fuel, primary system leakage, primary to secondary system leakage, steam generator blowdown and the performance of the various waste treatment systems.

Specification 2.4.3.a(1) limits the release rate of noble gases from the site so that the corresponding annual gamma and beta dose rate above background to an individual in an unrestricted area will not exceed 500 mrem to the total body or 3000 mrem to the skin in compliance with the limits of 10 CFR Part 20.

For Specification 2.4.3.a(1), gamma and beta dose factors for the individual noble gas radionuclides have been calculated for the plant gaseous release points and are provided in Table 2.4-3. The expressions used to calculate these dose factors are based on dose models derived in Section 7 of Meteorology and Atomic Energy-1968 and model techniques provided in Draft Regulatory Guide 1.AA.

Dose calculations have been made to determine the site boundary location with the highest anticipated dose rate from noble gases using onsite meteorological data and the dose expressions provided in Draft Regulatory Guide 1.AA. The dose expression considers the release point location, building wake effects, and the physical characteristics of the radionuclides.

The offsite location with the highest anticipated annual dose from released noble gases is 1600 meters in the North direction.

The release rate Specifications for a radioiodine and radioactive material in particulate form with half-lives greater than eight days are dependent on existing radionuclide pathways to man. The pathways which were examined for these Specifications are: 1) individual inhalation of airborne radionuclides, 2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, and 3) deposition onto grassy areas where milch animals graze with consumption of the milk by man. Methods for estimating doses to the thyroid via these pathways are described in Draft Regulatory Guide 1.AA. The offsite location with the highest anticipated thyroid dose rate from radioiodines and radioactive material in particulate form with half-lives greater than eight days was determined using onsite meteorological data and the expressions described in Draft Regulatory Guide 1.AA. Specification 2.4.3.a(2) limits the release rate of radioiodines and radioactive material in particulate form with half-lives greater than eight days so that the corresponding annual thyroid dose via the most restrictive pathway is less than 1500 mrem.

For radioiodines and radioactive material in particulate form with half-lives greater than eight days, the most restrictive location is a residence located 3,000 meters in the WSW direction (vent $\chi/Q=5.5\times10^{-7}$ sec/m^3).

Specification 2.4.3.b establishes upper offsite levels for the releases of noble gases and radioiodines and radioactive material in particulate form with half-lives greater than eight days at twice the design objective annual quantity during any calendar quarter, or four times the design objective annaul quantity during any period of 12 consecutive months. In addition to the limiting conditions for operation of Specifications 2.4.3.a and 2.4.3.b, the reporting requirements of 2.4.3.c provide that the cause shall be identified whenever the release of gaseous effluents exceeds one-half the design objective annual quantity during any calendar quarter and that the proposed program of action to reduce such release rates to the design objectives shall be described.

Specification 2.4.3.d requires that suitable equipment to monitor and control the radioactive gaseous releases are operating during any period these releases are taking place.

Specification 2.4.3.e limits the maximum quantity of radioactive gas that can be contained in a waste gas storage tank. The calculation of this quantity should assume instantaneous ground release, a X/Q based 5 percent meteorology, the average gross energy is 0.19 Mev per disintegration (considering Xe-133 to be the principal emitter) and exposure occurring at the minimum site boundary radius using a semiinfinite cloud model. The calculated quantity will limit the offsite dose above background to 0.5 rem or less, consistent with Commission guidelines.

Specification 2.4.3.f provides for reporting release events which, while below the limits of 10 CFR Part 20, could result in releases higher than the design objectives.

The sampling and monitoring requirements given under Specification 2.4.4 provide assurance that radioactive materials released in gaseous waste effluents are properly controlled and monitored in conformance with the requirements of Design Criteria 60 and 64. These requirements provide the data for the licensee and the Commission to evaluate the plant's performance relative to radioactive waste effluents released to the environment. Reports on the quantities of radioactive materials released in gaseous effluents are furnished to the Commission on the basis of Section 5.6.1 of these Technical Specifications. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

Specification 2.4.4.b excludes monitoring the turbine building ventilation exhaust since this release is expected to be a negligible release point. Many PWR reactors do not have turbine building enclosures. To be consistent in this requirement for all PWR reactors, the monitoring of gaseous releases from turbine buildings is not required.

2.4.5 Solid Waste Handling and Disposal

a. Measurements shall be made to determine or estimate the total curie quantity and principle radionuclide composition of all radioactive solid waste shipped offsite.

b. Reports of the radioactive solid waste shipments, volumes, principle radionuclides, and total curie quantity, shall be submitted in accordance with Section 5.6.1.

Bases

The requirements for solid radioactive waste handling and disposal given under Specification 2.4.5 provide assurance that solid radioactive materials stored at the plant and shipped offsite are packaged in conformance with 10 CFR Part 20, 10 CFR Part 71, and 49 CFR Parts 170-178.

3.0 ENVIRONMENTAL SURVEILLANCE

3.1 Non-Radiological Surveillance

3.1.A ABIOTIC

3.1.A.1 Biocides

Objective

The purpose of this surveillance is to monitor Total Residual Chlorine in the discharge canal to insure that no adverse impact on the environment is occurring.

Specification

Total Residual Chlorine shall be monitored in the discharge canal on a weekly basis while a condenser section is being chlorinated. See Section 2.2.1 for limiting conditions.

Reporting Requirement

Total Residual Chlorine concentration shall be reported in the Annual Environmental Monitoring Report.

3.1.A.2 Heavy Metals

Objective

The purpose of this study is to monitor heavy metals concentrations in the intake and discharge canals to detect any measurable increase in heavy metals.

Specification

Grab samples shall be taken on a monthly basis at the intake and discharge canals and analyzed for Mercury, Arsenic, Chromium, Copper, Iron, Lead, Nickel, and Zinc.

Reporting Requirement

Concentrations shall be reported in the Annual Environmental Monitoring Report.

3.1.A.3 pH

Objective

The purpose of this surveillance is to monitor pH in the receiving waters to insure that pH is not being raised or lowered from the specified limits, in order to prevent an adverse environmental impact.

Specification

pH shall be monitored daily using grab samples or a recorder in the discharge canal. See Section 2.2.2 for limiting conditions.

Reporting Requirements

pH measurements shall be reported in the Annual Environmental Monitoring Report.

3.1.A.4 Dissolved Oxygen

Objective

The purpose of this surveillance is to monitor dissolved oxygen (DO).

Specification

DO shall be monitored weekly, using grab samples, in the intake and discharge canals.

Reporting Requirements

Concentrations shall be reported in the Annual Environmental Monitoring Report.

3.1.A.5 Salinity

Objective

The purpose of this specification is to measure salinity concentrations in receiving waters.

Specification

Salinity shall be monitored by grab samples on a weekly basis in the discharge canal.

Reporting Requirements

Salinity concentrations shall be reported in the Annual Environmental Monitoring Report.

3.1.A.6 Temperature

Objective

To provide temperature data to limit thermal stress to the aquatic ecosystem.

Specification

A continuous temperature measurement system shall monitor circulating water temperature at the intake to Unit 1 and in the discharge canal. Both intake and discharge water temperature monitors shall have an accuracy of ±2°F. Signals shall be transmitted to the control room and displayed. The system shall have an alarm function to alert the control room operator of circulating water temperatures being at the maximum allowable limit.

A back-up system shall also be operable to monitor temperatures whenever the primary system fails. The back-up system does not have to transmit temperatures to the control room. Its overall accuracy shall also be ±2°F.

The maximum discharge temperature limitations shall be as described in Section 2.1.1.

In order to demonstrate compliance with the temperature rise limitations outside the zone of mixing, infrared aerial photography shall be employed, along with field measurements for ground truth. Four flights shall be scheduled during the first year of operation of Unit No. 1 after the unit is available for loading above 80% power level. Flights shall be spaced at approximately three month intervals, weather permitting, when the unit is operating at a power level of 80% or greater.

To demonstrate compliance with the temperature rise limitations within the zone of mixing, two self-contained recording thermographs shall be used. One thermograph shall be located at the surface of the water, at the point of maximum surface temperature of the Unit No. 1 discharge. This point has been determined by previous modeling studies. A second thermograph shall be located at the surface near the intake velocity cap of the Unit No. 1 to determine ambient temperature. These thermographs shall have a sensitivity of 0.5°F in a range from 40°F to 100°F.

Reporting Requirement

Results of this thermal monitoring program shall be summarized in the Annual Environmental Monitoring Report.

3.1.B BIOTIC

Objective

To determine the effects of plant operation on the planktonic, nektonic, and benthic populations of the Atlantic Ocean near the discharge during plant operation.

Specification

The biological conditions shall be assessed, 1) in terms of abundance and compositions of the marine biotic community, and 2) the relationship between certain chemical and physical properties of the waters and the character of the biological community.

The five sampling locations established during a pre-operational baseline biology program will be utilized for plankton, trawl, and benthic collections. The sampling schedule will be as follows:

a. Benthic Organisms - Benthic organisms will be collected quarterly and inventoried as to type and abundance of major taxonomic groups present.

b. Plankton - Plankton samples will be collected monthly. Both zooplankton and phytoplankton species will be identified as to kind and abundance. Chlorophyll "a" analysis will be performed as a measure of primary productivity.

c. Nektonic Organisms - Samples will be collected monthly by trawling, seining, or other suitable method. Types and numbers of organisms present will be determined, including species of migratory fish of commercial and sports fisheries value such as blue fish and mackerel.

d. Macrophytes - Macroscopic aquatic vegetation will be collected quarterly and identified as to species and abundance.

e. Water Quality - Analysis will be made on water samples taken at bottom, mid-depth, and surface levels at the same time as the biotic samples are collected. Parameters studied will be temperature, salinity, dissolved oxygen content, turbidity, and selected nutrients.

f. Migratory Sea Turtles - The species, numbers, and nesting characteristics of sea turtles that migrate in from the sea and nest along the east coast of Florida will be determined on the FPL shoreline property and selected adjacent control areas in 1975 and 1977. A study shall be conducted to determine the effects of the discharge thermal plume on turtle nesting patterns and turtle hatchling migration. In addition, control studies on temperature stress, hatching, and rearing factors will be conducted using turtle eggs from displaced nests.

Based on the data obtained, predictions will be made on the impact of the plant's operation on baseline biological conditions and current uses of the waters.

Florida Power and Light will review the data after two years of plant operation. If effects attributable to the plant are found acceptable, the results shall be reviewed by NRC to determine if the biotic program, or any portion thereof, should be terminated.

Reporting Requirement

Results of the biological program shall be reported in the Annual Environmental Monitoring Report.

3.2 RADIOLOGICAL ENVIRONMENTAL MONITORING

Objective

The Operational Radiological Environmental Surveillance Program is conducted to measure radiation levels and radioactivity in the environs, and to assist in verifying any projected or anticipated radioactivity release resulting from plant operations which could bring about public exposure to radiation.

Specifications

3.2.a Environmental samples shall be collected at the designated locations shown in Table 3.2-1 and Figures 3.2-1 and 3.2-2.

3.2.b The criteria for the type and the number of samples to be collected at a given sampling location, the frequency of collection, and the type and frequency of radioactivity analysis to be completed on the collected samples shall be as shown in Table 3.2-2.

Direct radiation shall be measured by thermoluminescence dosimetry (TLD) at locations shown in Table 3.2-1 and Figures 3.2-1 and 3.2-2. The system shall be capable of measuring 26 mrem/year with a precision of $\pm$10% at the 95% confidence level based on a quarterly collection frequency.

3.2.c The radiation detection capabilities of the radioanalytical methods used shall be as shown in Table 3.2-3.

3.2.d A census of gardens producing fresh leafy vegetation for human consumption shall be conducted near the end of the growing season to determine their location with respect to the plant site. This census is limited to gardens having an area of 500 ft^2 or more, and shall be conducted under the following conditions:

1. Within a 1 mile radius of the plant site, enumerated by door-to-door or equivalent counting technique.

2. If no milk-producing animals are located in the vicinity of the site, as determined by Specification 3.2.e below, the census described in item 1, above, shall be extended to a distance of 5 miles from the site.

3. If this census reveals the existence of a garden at a location yielding a calculated thyroid dose greater than that from a previously sampled garden, the new location shall replace the garden previously having the maximum iodine concentration. Also, any location from which fresh leafy vegetables can no longer be obtained may be dropped from the

TABLE 3.2-1

ST. LUCIE PLANT

OPERATIONAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE PROGRAM

SAMPLING LOCATIONS AND VECTORS SAMPLED

Station No.	Description	Bearing*	Distance*	Vector Sampled
H03	Meadowbrook Dairy, Glades Cut-off Road, St. Lucie County	260°	22.526 km (14.00 mi)	Milk
H08	Florida Power & Light Company Substation White City, Weatherby Road west of U. S. 1	293°	9.170 km (5.7 mi)	Soil, Air Particulates & Iodine, Direct Radiation
H09	Florida Power & Light Company Substation west of U.S. 1, just south of St. Lucie County Line	196°	11.745 km (7.30 mi)	Soil, Air Particulates & Iodine, Direct Radiation
H10	Indian River Field Laboratory, University of Florida, west of SR 713	300°	19.308 km (12.00 mi)	Food Crops (Citrus), Air Particulates & Iodine, Direct Radiation, Soil
H11	City of Ft. Pierce Water System - Collected at St. Lucie County Health Department, Ave., "C", Ft. Pierce	323°	14.480 km (9 mi)	Potable Water (Well) - City of Ft. Pierce
H12	Florida Power & Light Company Substation SR 76 west of U.S. 1, Stuart, Martin County	180°	19.308 km (12.00 mi)	Potable Water (Well) - City of Stuart, Air Particulates & Iodine, Direct Radiation
H13	On Site, Point north of Big Mud Creek at Indian River	312°	1021 m (0.63 mi)	Surface Water, Bottom Sediment (Estaurine)

TABLE 3.2-1 (Continued)

Station No.	Description	Bearing*	Distance*	Vector Sampled
H14	Employees Parking Lot, southeast of Containment	160°	503 m (0.31 mi)	Air Particulates & Iodine, Direct Radiation
H15	On Site, Beach near Discharge Structure	89°	808 m (0.50 mi)	Ocean Water & Bottom Sediment, Aquatic Biota
H16	Beach (ocean) opposite Blind Creek	31°	1509 m (0.94 mi)	Ocean - Bottom Sediment, Beach Sand
H19	On Site, Beach south of Intake Canal	161°	1494 m (0.90 mi)	Ocean - Bottom Sediment, Beach Sand
H22	Lentz Groves, U.S. 1	210°	8.849 km (5.50 mi)	Food Crop (Citrus)
H23	Montauk Groves, U.S. 1, south of Easy Street	270°	7.562 km (4.70 mi)	Food Crop (Citrus)
H24	Poster Groves, U.S. 1, north of Tumblin Kling Road	300°	8.608 km (5.35 mi)	Food Crop (Citrus)
H25	Childs Groves, Bell Avenue, west of Sunrise Blvd.	297°	11.263 km (7.00 mi)	Food Crop (Citrus)
H26	Wouters Groves, west of SR 713 on Immokola Road	314°	21.720 km (13.50 mi)	Food Crop (Citrus)
H30	Residence, 7609 Indian River Drive	245°	3.218 km (2.00 mi)	Ground Water (Well), Soil, Air Particulates & Iodine, Direct Radiation

TABLE 3.2-1 (Continued)

Station No.	Description	Bearing*	Distance*	Vector Sampled
H31	North Port St. Lucie Water System, Prima Vista Blvd.	250°	10.619 km (6.60 mi)	Potable Water (Well) - Port St. Lucie
H32	Department of Health and Rehabilitative Services Entomology Laboratory, East of U.S. 1, Vero Beach	338°	30.571 km (19.00 mi)	Aquatic Biota, Ocean Water & Bottom Sediment, Air Particulates & Iodine, Soil, Direct Radiation, Beach Sand
H33	On Site, between Canals, east of AIA	138°	945 m (0.59 mi)	Air Particulates & Iodine, Direct Radiation
H34	On Site, Meteorological Tower	27°	762 m (0.47 mi)	Air Particulates & Iodine, Direct Radiation
H36	On Site, Discharge Canal west of AIA	101°	305 m (0.19 mi)	Surface Water, Bottom Sediment
H39	Vista Royal Condominium, 1 mile north of H32, east of U.S. 1, Vero Beach	338°	32.180 km (20.00 mi)	Food Crop (Citrus)
H40	Davis Dairy, Military Trail, west of Boynton Beach, Palm Beach County	172°	89.770 km (55.77 mi)	Milk

*Bearings and distances from the center of Generating Stations.

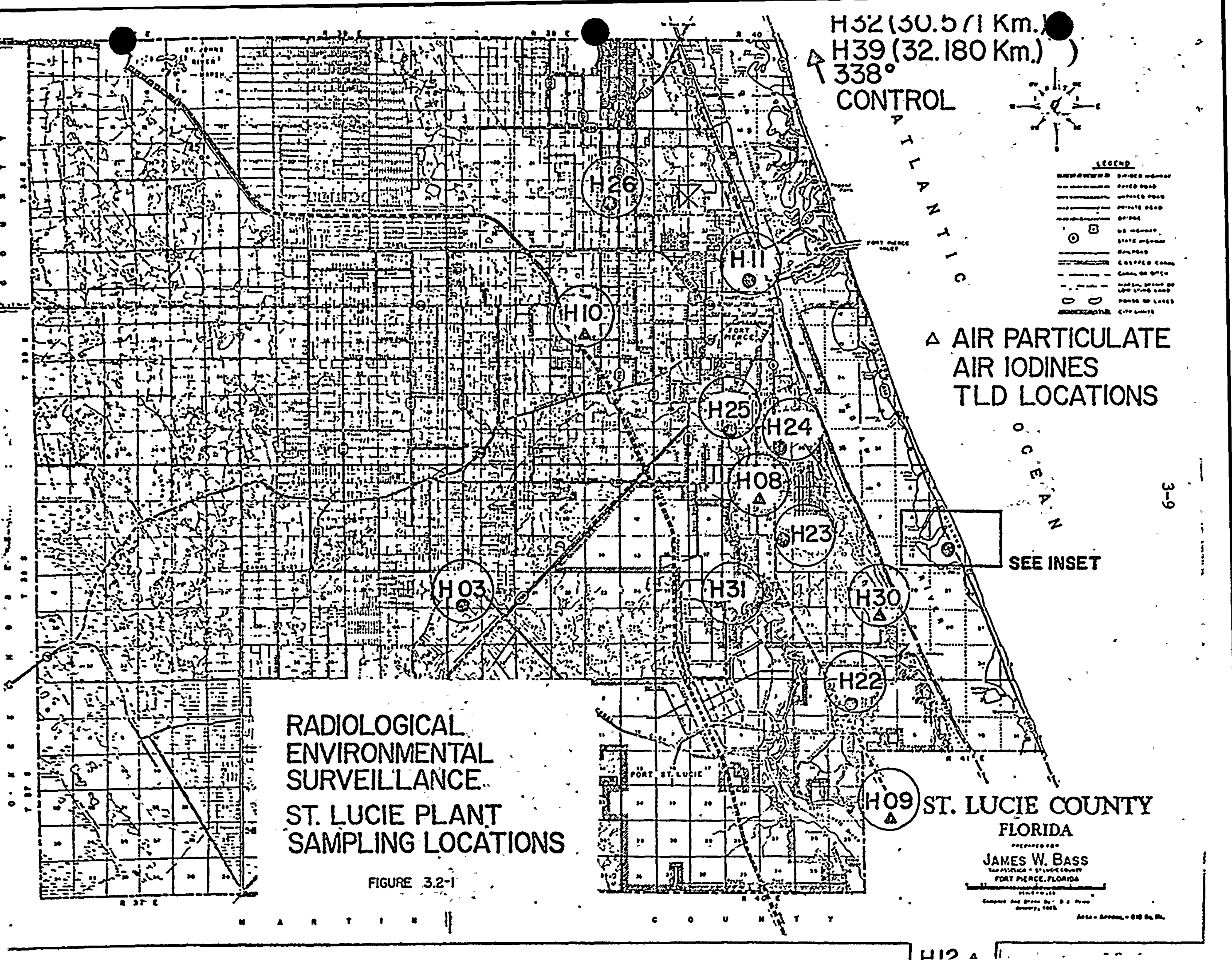
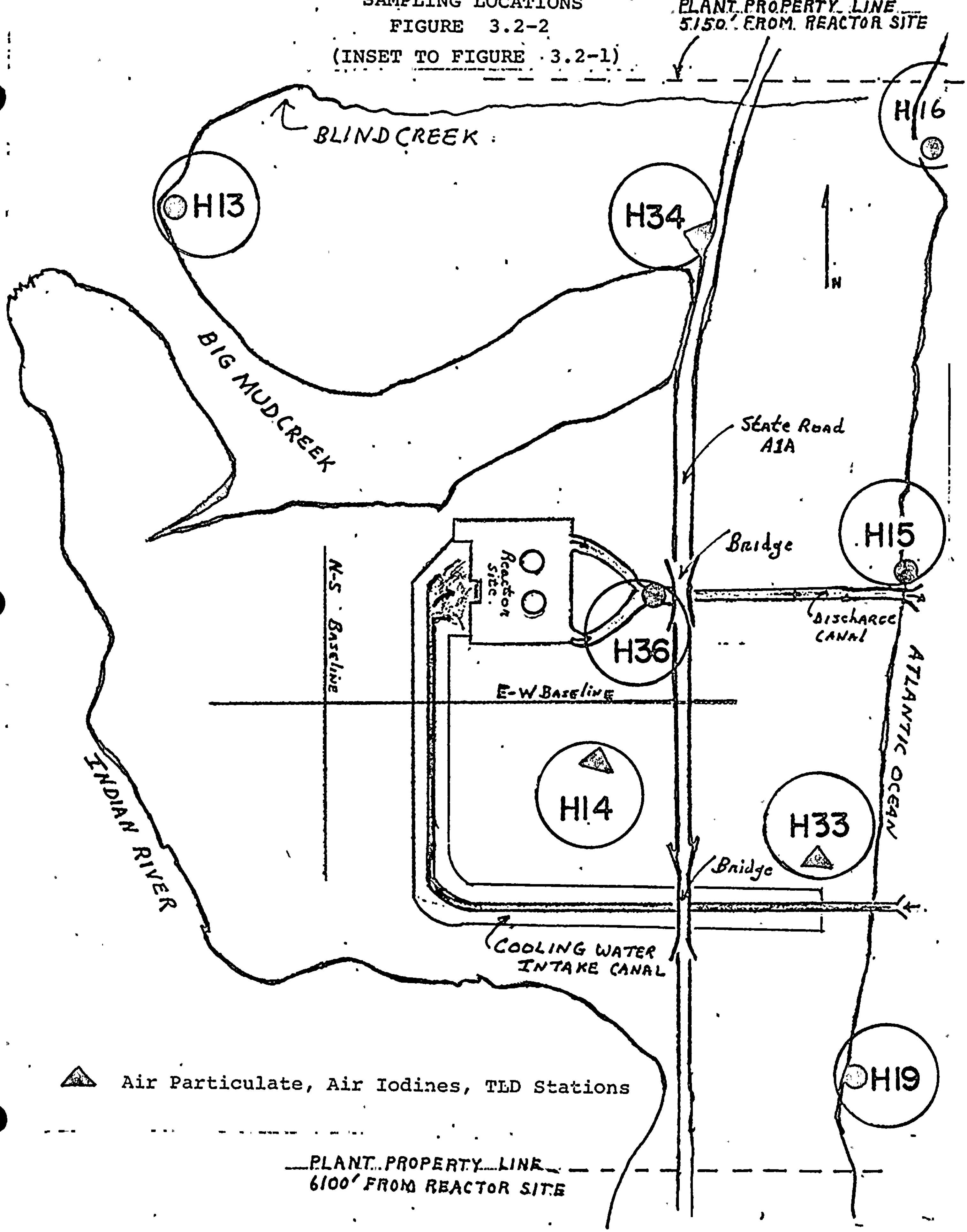

H32 (30.571 Km.)
H39 (32.180 Km.)
338°
CONTROL
ATLANTIC OCEAN
H26
H11
H10
H25
H24
H08
H23
H03
H31
H30
H22
H09
SEE INSET
FORT PIERCE
PORT ST. LUCIE
△ AIR PARTICULATE
AIR IODINES
TLD LOCATIONS
LEGEND
RADIOLOGICAL
ENVIRONMENTAL
SURVEILLANCE
ST. LUCIE PLANT
SAMPLING LOCATIONS
ST. LUCIE COUNTY
FLORIDA
JAMES W. BASS
FORT PIERCE, FLORIDA
MARTIN COUNTY

FIGURE 3.2-1

ST. LUCIE PLANT
SAMPLING LOCATIONS
FIGURE 3.2-2
(INSET TO FIGURE 3.2-1)



Air Particulate, Air Iodines, TLD Stations

TABLE 3.2.2 SHEET 1

OPERATIONAL ENVIRONMENTAL RADIOLOGICAL SURVEILLANCE PROGRAM

ST. LUCIE PLANT

Exposure Pathway and/or Sample	Criteria and Sampling Locations	Collection Frequency	Type and Rrequency of Analysis
1. AIR			
1.1 Particulate and Iodine	Comparison on-site versus off-site & reference locations: 3 locations on-site, north, east, & southeast of the plant: H 34, H 14, H 33; 5 locations off-site within a radius of 10 miles of plant: H 08, H 09, H 10, H 12, H 30, and 1 control location: H32	Weekly	Gross Beta Gamma spectral analysis of monthly composite Radioactive Iodine Sr-89 & 90 (Quarterly Composite)
1.2 Direct Radiation	Comparison of on-site versus off-site & reference locations: 3 locations on-site, north, east, & southeast of the plant: H 34, H 14, H 33; 5 locations off-site within a radius of 10 miles of plant: H 08, H 09, H 10, H 12, H 30, and 1 control location: H32	Quarterly	Determine direct radiation exposure by TLD readout (mean of 2 TLDs)
2. WATER			
2.1 Surface Water			
2.1.1 Discharge Canal	1 location, west of AIA: H36	Monthly	Gamma spectral analysis Tritium (Quarterly Composite) Sr-89 & 90 (Quarterly Composite)
2.1.2 Ocean	2 locations; H15 & H32 (Control)	Monthly	Gamma spectral analysis Tritium (Quarterly Composite) Sr-89 & 90 (Quarterly Composite)
2.1.3 Estuarine	1 location; Big Mud Creek: H13	Quarterly	Gamma spectral analysis Tritium
2.2 Ground Water (well)	1 location; Residence, 7609 Indian River Drive: H30	Semi-annually	Gamma Spectral Analysis Gross Beta Tritium

TABLE 3.2.2 SHEET 2

OPERATIONAL ENVIRONMENTAL RADIOLOGICAL SURVEILLANCE PROGRAM

ST. LUCIE PLANT

Exposure Pathway and/or Sample	Criteria and Sampling Locations	Collection Frequency	Type and Frequency of Analysis
2. WATER (cont'd)			
2.3 Potable Water (wells)	1 location, City of Ft. Pierce, drinking water supply, H 11 1 location, City of Stuart, drinking water supply, H 12 1 location, Port St. Lucie, drinking water supply, H 31	Quarterly	Gamma spectral analysis Gross Beta Tritium
3. BOTTOM SEDIMENT			
3.1 Discharge Canal	1 location, west of AIA: H36	Semi-annually	Gamma spectral analysis Sr-90
3.2 Ocean	1 location, beach west of discharge structure: H15 1 location, offshore, 1 mile north of discharges: H16 1 location, offshore, 1 mile south of discharge: H19 1 location, offshore, Vero Beach: H32 (Control)	Semi-annually	Gamma spectral analysis Sr-90
3.3 Beach (sand)	1 location, east of Blind Creek, 1 mile north of discharge: H16 1 location, near intake, 1 mile south of discharge: H19 1 location, Vero Beach: H32 (Control)	Semi-annually	Gamma spectral analysis Sr-90
3.4 Estuarine	1 location, Big Mud Creek: H13	Semi-annually	Gamma spectral analysis
4. AQUATIC BIOTA			
4.1 Crustacea (Lobster or crab or shrimp)	1 location, vicinity of discharge structure: H15 1 location, Vero Beach: H32 (Control)	Semi-annually	Gamma spectral analysis

TABLE 3.2.2 SHEET 3

OPERATIONAL ENVIRONMENTAL RADIOLOGICAL SURVEILLANCE PROGRAM

ST. LUCIE PLANT

Exposure Pathway and/or Sample	Criteria and Sampling Locations	Collection Frequency	Type and Frequency of Analysis
4. AQUATIC BIOTA (cont'd)			
4.2 Fish			
4.2.1 Carnivores	1 location, vicinity of discharge structure: H 15 1 location, Vero Beach: H32 (Control)	Semi-annually	Gamma spectral analysis Sr-89 & 90
4.2.2 Herbivores	1 location, vicinity of discharge structure: H15 1 location, Vero Beach: H32 (Control)	Semi-annually	Gamma spectral analysis Sr-89 & 90
5. TERRESTRIAL			
5.1 Milk	1 location within 15 mile radius of plant and in the prevailing wind direction from the plant: H03	Semi-monthly	Gamma spectral analysis Sr-89 & 90 I-131
	1 location, 55.77 mi south of the plant, Palm Beach County H40 (Control)	Monthly	Gamma spectral analysis Sr-89 & 90 I-131
	Dairy herd census	Semi-annually	
5.2 Biota			
5.2.1 Food Crop (Citrus)	6 locations, H10, H22, H23, H24, H25, H26	Harvest Time	Gamma spectral analysis Sr-89 & 90
	1 location, Vero Beach: H39 (control)	Harvest Time	Gamma spectral analysis Sr-89 & 90
5.2.2 Food Crop (Edible Leafy vegetation)	1 location as determined by garden census (Specification 3.2.d)	Harvest Time	Gamma spectral analysis I-131
5.3 Soil	5 locations within a 15 mile radius of plant: H03, H08, H09, H10, H30. 1 location, Vero Beach: H32 (Control)	Once per 3-year period	Gamma spectral analysis Sr-90

TABLE 3.2-3

St. Lucie Plant: Detection Capabilities for Environmental Sample Analysis

Analysis	Media Detection Capabilities*					
	Water (pCi/l)	Airborne Particulates or Gas (pCi/m^3)	Fish, Meat or Poultry (pCi/kg,wet)	Milk (pCi/l)	Vegetation (pCi/kg,wet)	Soil (pCi/kg,dry)
Gross beta	0.8	0.002				
^{3}H	199.0					
^{54}Mn	6.0		17.0			
^{59}Fe	-		5.0			
^{58}Co	7.0		19.0			
^{60}Co	7.0		20.0			
^{65}Zn	14.0		39.0			
^{89}Sr	1.6	0.005	8.0	2.0		
^{90}Sr	0.8	0.002	4.0	1.0		10.0
^{95}Zr-Nb	7.0					
^{131}I	7.6	0.008		0.5	16.0	
^{134}Cs	6.0	0.008	18.0	6.0		26.0
^{137}Cs	7.0	0.008	18.0	7.0		26.0
^{140}Ba-La	8.0	0.008				

*Nominal: LLD's (lower limit of detection) calculated as defined in HASL-300, Rev. 8/73, pp 08-01, 02, 03, at the 90% confidence level. The detection levels for radionuclides analyzed by gamma spectrometry will vary according to the number of radionuclides encountered in environmental samples.

surveillance program after notifying NRC in writing that such vegetables are no longer grown at that location.

3.2.e A census of animals producing milk for human consumption shall be conducted semiannually to determine their location and number with respect to the plant site. The census shall be conducted under the following conditions:

1. Within a 1 mile radius from the plant site or within a 15 mrem/year isodose line (as calculated using dose models presented in Regulatory Guide 1.42), whichever is larger, enumeration by a door-to-door or equivalent counting technique.

2. Within a 5 mile radius for cows and a 15 mile radius for goats, enumeration by using referenced information from county agricultural agents or other reliable sources.

If it is determined from the census that animals are present at a location which yields a calculated thyroid dose greater than that from previously sampled animals, the new location shall be added to the surveillance program as soon as practicable. The sampling location having the lowest calculated dose may be dropped from the surveillance program 3 months after sampling begins at the new location. Also, any location from which milk can no longer be obtained may be dropped from the surveillance program after notifying NRC in writing that milk-producing animals are no longer present at that location.

3.2.f Deviations from the required sampling schedule are permitted if specimens are not obtainable due to hazardous conditions, seasonal unavailability, or malfunction of automatic sampling equipment. In the latter case, every reasonable effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be described in the annual report.

A deviation of not greater than one week from the required frequency of analysis for gross beta activity, as shown in Table 3.2-2, is permitted if equipment failure delays the analyses. Every reasonable effort shall be made to effect expeditious equipment repair.

Reporting

All required reports from this Operational Radiological Environmental Surveillance Program shall be prepared and presented in the manner described in Section 5.6.1.B of these Environmental Technical Specifications.

Bases

The program is designed to determine existing radioactivity levels and to detect changes in radiation levels in the air, water and land environment which may be attributed to the operation of the plant. The methods,

procedures and techniques used were developed during the preoperational phase and have provided background measurements that will be used as a base for distinguishing significant changes in radioactivity in the site environs.

3.3 ONSITE METEOROLOGICAL MONITORING

Objective

The objective of onsite meteorological monitoring is to adequately measure and document meteorological conditions at the site, specifically at heights above ground that are representative of atmospheric conditions that exist at all effluent release points.

Specification

The onsite meteorological monitoring program shall conform to the recommendations and intent of Regulatory Guide 1.23, Onsite Meteorological Programs, and include instruments to sense wind speed and direction at 33 ft and 190 ft, vertical temperature gradient between 33 ft and 200 ft, and ambient dry bulb and dewpoint temperatures at 33 ft. The location of the meteorological tower shall be located approximately 2400 ft north of the reactor complex.

Reporting Requirements

Meteorological data shall be summarized and reported in a format consistent with the recommendations of Regulatory Guides 1.21 and 1.23. Summaries of data and observations shall be available to the U. S. Nuclear Regulatory Commission upon request. If the outage of any meteorological instrument(s) exceeds seven consecutive days, the total outage time and dates of outage, the cause of the outage, and the instrument(s) involved shall be reported within 30 days of the initial time of the outage to the U. S. Nuclear Regulatory Commission, Office of Inspection and Enforcement, with a copy to the Office of Nuclear Reactor Regulation, Division of Technical Review. Any modifications to the meteorological monitoring program, as described above, or alterations of the area near the meteorological tower that would interfere with the measurement of meteorological conditions representative of the site, shall have the written approval of the U. S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, prior to the initiation of the modification or alteration.

Bases

The collection of meteorological data at the plant site will provide information which will be used to develop atmospheric diffusion parameters to estimate potential radiation doses to the public resulting from actual routine or accidental releases of radioactive materials to the atmosphere.

4.0 SPECIAL SURVEILLANCE AND SPECIAL STUDY ACTIVITIES

4.1 Entrainment of Aquatic Organisms

Objective

The purpose of this study is to assess the effects on planktonic organisms of passage through the plant condensers. Specialists in the biological sub-disciplines of zooplankton and ichthyology will perform appropriate portions of this study. Figures obtained for the intake and discharge canals will be compared to data collected at a control station.

Specification

Samples shall be collected from the intake and discharge canals and a control station at monthly intervals when the unit is in operation to identify the organisms involved, and to attempt to quantify how many of each organism are potentially affected. Biomass measurements, numbers of eggs collected, and numbers and identification of larvae - to the level of major taxonomic groups, if possible - shall be performed. Present "state-of-the-art" information shall be used to attempt to quantify the mortality of the organisms due to entrainment. This program shall determine the seasonal abundance of fish eggs and larvae.

Reporting Requirements

Results of this study shall be summarized in the Annual Environmental Monitoring Report. If, at the end of two years, no significant problem is evident, an option to formally delete this portion of the Technical Specifications may be initiated.

4.2 Impingement of Aquatic Organisms

Objective

The purpose of this study is to assess the impingement of aquatic organisms on intake screens and the environmental impact of the impingement.

Specification

Intake screens washings shall be examined for a consecutive twenty-four hour period, twice a week whenever the Unit 1 circulating water pumps are operating. The collected washings shall be analyzed for the species present, number of each individual species caught, total biomass of each species, and the average size of the individuals caught.

Reporting Requirements

Data collected shall be analyzed monthly for the first year of operation and a report sent to the NRC within 45 days of each monthly period. After the first year of operation, the data shall be analyzed every six months, and the results summarized in the Annual Environmental Monitoring Report.

4.3 Minimum Effective Chlorine Usage

Objective

The purpose of this study is to determine the minimum amount of chlorine necessary which will afford adequate protection to the condenser while avoiding unnecessary discharge of chlorine to the environment.

Specification

A program shall be initiated after Unit 1 has initially reached 75% power level. The initial chlorine injection rate shall be determined based on preoperational data, previous experience, and laboratory chlorine demand tests. After reaching a power plateau above 75% power, a controlled incremental reduction of the chlorine injection rate shall be implemented. Condenser fouling shall be monitored in coordination with chlorine reduction.

Reporting Requirements

The results of this study shall be summarized in the Annual Environmental Monitoring Report. When the minimum level of chlorine usage, as determined by the study, has been reached, a proposal shall be submitted to the NRC to terminate the study.

5.0 ADMINISTRATIVE CONTROLS

The purpose of this section is to describe the administrative and management controls necessary to provide continuing protection to the environment, and to implement the environmental technical specifications (ETS).

5.1 Responsibility

The Vice President of the Environmental Department has the ultimate responsibility for the implementation of the ETS. He may delegate to other departments and/or organizations the work of establishing and executing portions of the ETS, but shall retain responsibility thereof.

The Environmental Department is responsible for executing the non-radiological biotic and special studies sections of the ETS. The Vice President of Power Resources is responsible for executing the non-radiological abiotic, radioactive effluents, and the Radiological Environmental Surveillance sections.

The Plant Quality Control group shall be responsible for the day-to-day verification of compliance with the ETS. The Manager of Quality Assurance shall be responsible for periodic audits, conducted according to the corporate Quality Assurance program, to insure compliance with the ETS.

5.2 Organization

The corporate organization involved in environmental matters is depicted in Figure 5.2-1.

5.3 Review and Audit

5.3.1 Review of implementation of the ETS shall be made by the Company Environmental Review Group (CERG) or by the Plant Nuclear Safety Committee (PNSC). Secondary reviews shall be made by the Company Nuclear Review Board (CNRB).

5.3.2 PNSC and CNRB membership and responsibilities are described in Appendix A, Technical Specifications.

5.3.3 Company Environmental Review Group (CERG)

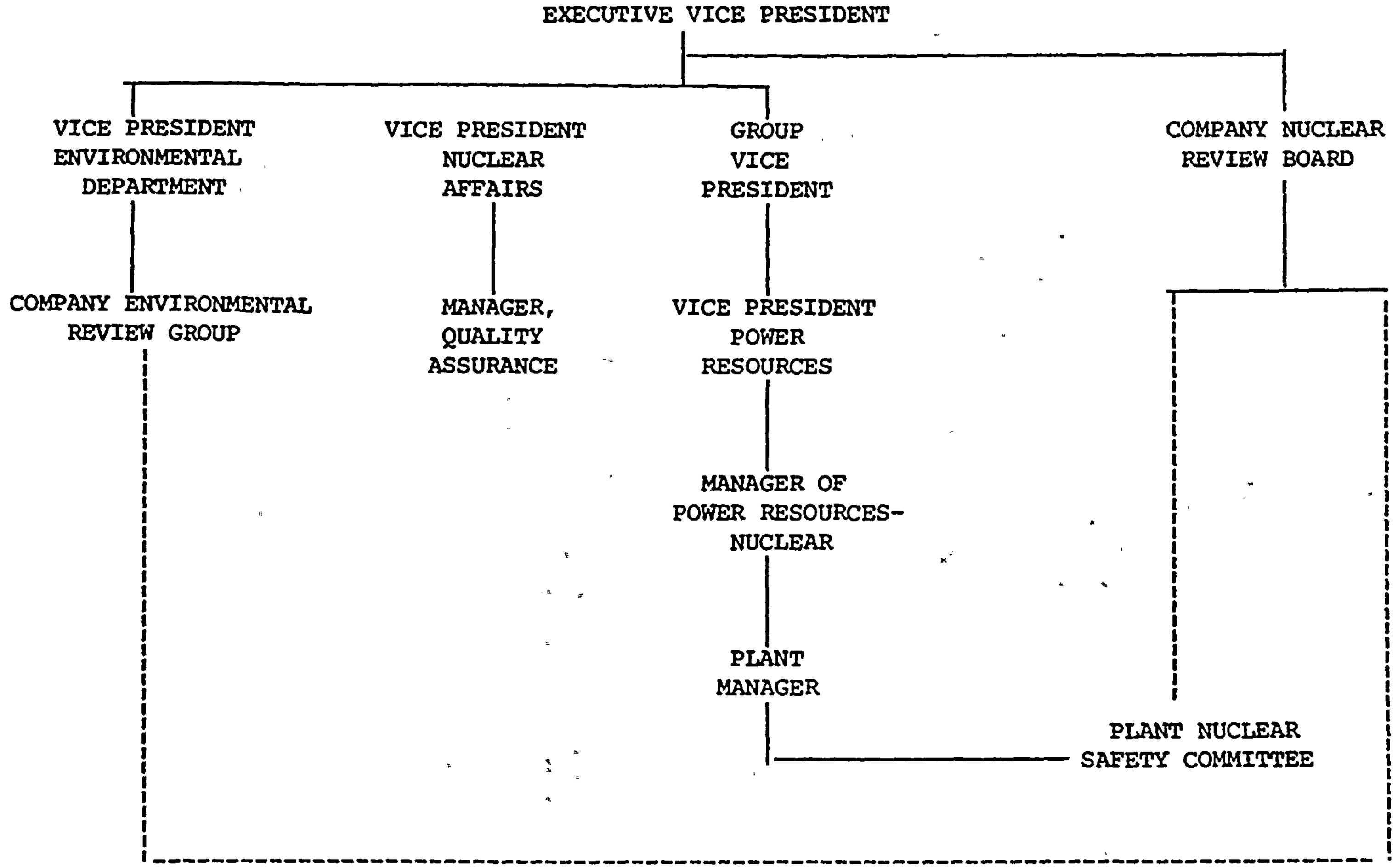
A. Function

The Company Environmental Review Group (CERG) shall function to advise the Vice President, Environmental Department on all matters related to environmental quality.

B. Membership

1. Manager, Environmental Engineering - Chairman

2. Manager, Environmental Affairs



FLORIDA POWER & LIGHT COMPANY
CORPORATE ORGANIZATION - ENVIRONMENTAL AFFAIRS
FIGURE 5.2-1

2-17-75

——— AUTHORITY

------- COMMUNICATION

3. Senior Environmental Engineer, Environmental Engineering

4. Power Resources Test Group Supervisor

5. Environmental Department Life Scientist

6. Environmental Department Senior Project Coordinator

7. Plant Supervisor (Plant Involved)

8. Power Resources Administrative Assistant - Nuclear

C. Alternates

Alternate members shall be appointed in writing by the CERG Chairman. No more than two alternates shall participate in CERG activities at any one time.

D. Meeting Frequency

The CERG shall meet at least semiannually and as convened by the CERG Chairman or designated acting Chairman.

E. Quorum

A quorum of the CERG shall consist of the Chairman, or designated acting Chairman and three members including alternates.

F. Responsibilities

1. Review of all Environmental Department procedures required by Environmental Technical Specifications and changes thereto. Review of any proposed procedures or changes thereto as determined by the Plant Manager to affect the environment.

2. Review results of the environmental monitoring programs prior to their submittal to the NRC.

3. Review of all proposed test and experiments as determined by the Plant Manager to affect the environment.

4. Review of all proposed changes to the Environmental Technical Specifications.

5. Review of all proposed changes or modifications to plant systems or equipment as determined by the Plant Manager to affect the environment.

6. Review of investigation of violations of the Environmental Technical Specifications.

7. Performance of special reviews and investigations and reports thereon as required by the Chairman of the Company Nuclear Review Board.

G. Authority

The Company Environmental Review Group shall:

- Recommend to the Vice President, Environmental Department, written approval or disapproval of the items considered under F.1 through F.5 above.

H. Records

The Company Environmental Review Group shall maintain written minutes of each meeting and copies shall be provided to the Vice President, Environmental Department, Vice President, Power Resources, and the Chairman of the Company Nuclear Review Board.

5.3.4 Periodic audits concerning the implementation of the ETS shall be made as provided in the Quality Assurance Manual.

5.4 Action to be Taken if a Limiting Condition is Exceeded

5.4.1 When a Limiting Condition is exceeded, action shall be taken as permitted by the applicable specification until the condition can be met.

5.4.2 Exceeding a Limiting Condition shall be investigated by the Company Environmental Review Group or by the Plant Nuclear Safety Committee.

5.4.3 All reviews and actions taken, with reasons therefor, shall be recorded and maintained as part of the permanent records.

5.4.4 Each instance whereby a Limiting Condition is exceeded shall be reported to the Company Nuclear Review Board.

5.5.5 A report for each occurrence shall be prepared as specified in Section 5.6.2.

5.5 Procedures

5.5.1 Detailed written procedures, including applicable check lists and instructions, shall be prepared and followed for activities involved in carrying out the environmental technical specifications. Procedures shall include sampling, data recording and storage, instrument calibration, measurements and analyses, and actions to be taken when limits are exceeded. Testing frequency of any alarms shall be included.

5.5.2 Plant operating procedures shall include provisions to ensure that plant systems and components are operated in compliance with the environmental technical specifications.

5.6 Reporting Requirements

5.6.1 Routine Reports

5.6.1.a Annual Non-Radiological Environmental Monitoring Report

A report on the environmental surveillance programs for the previous 12 months of operation shall be submitted to the Director of the Regional Office of Inspection and Enforcement with a copy to the Director of the Office of Inspection and Enforcement as a separate document within 90 days after January 1 of each year. In the event that some of the results are not available within the 90 day period, the report shall be submitted noting and explaining the missing results. The missing data shall be submitted as soon as possible in a supplementary report. The period of the first report shall begin with the date of initial criticality. The report shall include summaries and interpretations of the results of the non-radiological environmental surveillance activities (Section 3.0) and the environmental monitoring programs required by Limiting Conditions for Operation. This should also include a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports, and an assessment of the observed impacts of the plant operation on the environment. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee shall provide an analysis of the problem and a proposed course of action to alleviate the problem.

5.6.1.b Annual Radiological Environmental Monitoring Report

A report on the radiological environmental surveillance programs for the previous 12 months of operation shall be submitted to the Director of the NRC Regional Office (with a copy to the Director, Office of Nuclear Reactor Regulation) as a separate document within 90 days after January 1 of each year. The period of the first report shall begin with the date of initial criticality. The reports shall include summaries, interpretations, and statistical evaluation of the results of the radiological environmental surveillance activities for the report period, including a comparison with preoperational studies, operational controls (as appropriate), and previous environmental surveillance reports and an assessment of the observed impacts of the plant operation on the environment. The reports shall also include the results of land use censuses required by the specifications. If harmful effects or evidence of irreversible damage are detected by the monitoring, the licensee shall provide an analysis of the problem and a proposed course of action to alleviate the problem.

Results of all radiological environmental samples taken shall be summarized on an annual basis in a format similar to that indicated in Table 5.6.1-F. In the event that some results are not available within the 90-day period, 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.

TABLE 5.6.1-F

FORMAT FOR ENVIRONMENTAL RADIOLOCICAL MONITORING PROGRAM SUMMARY

NAME OF FACILITY ST. LUCIE PLANT, UNIT 1 DOCKET NO. 50-335

LOCATION OF FACILITY ST. LUCIE COUNTY, FLORIDA REPORTING PERIOD ________________

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection[a] (LLD)	All Indicator Locations Mean (f)[b] Range[b]	Location with Highest Annual Mean		Control Locations Mean (f)[b] Range[b]	Number of Nonroutine Reported Measurements[c]
				Name Distance and Direction	Mean (f)[b] Range[b]		
Air Particulates							
(pCi/m^3)	Gross β 416	0.003	0.08 (200/312) (0.05-2.0)	Middletown 5 miles 340°	0.10 (5/52) (0.08-2.0)	0.08 (8/104) (0.05-1.40)	1
	γ -Spec.32						
	^{137}Cs	0.003	0.05 (4/24) (0.03-0.13)	Smithville 2.5 miles 160°	0.08 (2/4) (0.03-013)	<LLD	4
	^{140}Ba	0.003	0.03 (2/24) (0.01-0.08)	Podunk 4.0 miles 270°	0.05 (2/4) (0.01-0.08)	0.02 (1/8)	1
	^{89}Sr 40	0.002	<LLD	-	-	<LLD	0
	^{90}Sr 40	0.0003	<LLD	-	-	<LLD	0
Fish							
pCi/kg (dry weight)	γ -Spec. 8						
	^{137}Cs	80	<LLD	-	<LLD	90 (1/4)	0
	^{134}Cs	80	<LLD	-	<LLD	<LLD	0
	^{60}Co	80	120 (3/4) (90-200)	River Mile 35 Podunk River	See column 4	<LLD	0

[a]Nominal Lower Limit of Detection (LLD) as defined in HASL-300 (Rev. 8/74) pp. D-08-01, 02, 03 at the 95% compliance level.

[b]Mean and range based upon detection measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses (f).

[c]Nonroutine reported measurements as defined in Section 5.6.2.b.

Example Data Presentation

5.6.1.c Semiannual Radioactive Effluent Release Report

A report on the radioactive discharges (Regulatory Guide 1.21, Rev. 1, June 1974) released from the site during the previous 6 months of operation shall include the following:

Analyses of Effluent releases shall be summarized on a quarterly basis and reported in a format similar to Tables 5.6.1-A, B, C, and D.

Supplemental information shall be included covering topics similar to those itemized in Data Sheet 5.6.1-1.

Abnormal releases should be handled as batch releases for accounting purposes.

Solid wastes shall be summarized on a quarterly basis and reported in a format similar to that of Table 5.6.2-E.

The following information should be reported for shipments of solid waste and irradiated fuel transported from the site during the report period:

1. The semiannual total quantity in cubic meters and the semiannual total radioactivity in curies for the categories or types of waste.

    a. Spent resins, filter sludges, evaporator bottoms;

    b. Dry compressible waste, contaminated equipment, etc.;

    c. Irradiated components, control rods, etc.;

    d. Other (furnish description).

2. An estimate of the total activity in the categories of waste in 1, above.

3. The disposition of solid waste shipments. (Identify the number of shipments, the mode of transport, and the destination.)

4. The disposition of irradiated fuel shipments. (Identify the number of shipments, the mode of transport, and the destination.)

5.6.2 Non-Routine Reports

5.6.2.a Non-Radioactive Effluent Reports

A report shall be submitted in the event that: a) a limiting condition is exceeded (as specified in Section 2.0 Limiting Conditions), or an unusual or important event occurs that causes a significant environmental impact, that affects potential environmental impact from plant operation, or that has high public or potential public interest concerning environmental impact from plant operation. Reports shall be submitted under one of the report schedules described below.

TABLE 5.6.1-1

EFFLUENT AND WASTE DISPOSAL

Supplemental Information

Facility ______________________ License ____________________

1. Regulatory Limits

   a. Fission and activation gases:
   b. Iodines:
   c. Particulates, half-lives >8 days:
   d. Liquid effluents:

2. Maximum Permissible Concentrations

   Provide the MPCs used in determining allowable release or concentrations.

   a. Fission and activation gases:
   b. Iodines:
   c. Particulates, half-lives >8 days:
   d. Liquid effluents:

3. Average Energy

   Provide the average ($\bar{E}$) of the radionuclide mixture in releases of fission and activation gases, if applicable.

4. Measurements and Approximations of Total Radioactivity

   Provide the methods used to measure or approximate the total radioactivity in effluents and the methods used to determine radionuclide composition.

   a. Fission and activation gases:
   b. Iodines:
   c. Particulates:
   d. Liquide effluents:

5. Batch Releases

   Provide the following information relating to batch releases of radioactive materials in liquid and gaseous effluents.

a. Liquid

1. Number of batch releases:
2. Total time period or batch releases:
3. Maximum time period for a batch release:
4. Average time period for batch releases:
5. Minimum time period for a batch release:
6. Average stream flow during periods of release of effluent into a flowing stream:

b. Gaseous

1. Number of batch releases:
2. Total time period for batch releases:
3. Maximum time period for a batch release:
4. Average time period for batch releases:
5. Minimum time period for a batch release:

6. Abnormal Releases

a. Liquid

1. Number of releases:
2. Total activity released:

b. Gaseous

1. Number of releases:
2. Total activity released:

TABLE 5.6.1-A

GASEOUS EFFLUENTS-SUMMATION OF ALL RELEASES

	Unit	Quarter	Quarter
A. Fission & activation gases			
1. Total release	Ci	. E	. E
2. Average release rate for period	μCi/sec	. E	. E
3. Percent of Technical specification limit	%	. E	. E
B. Iodines			
1. Total iodine-131	Ci	. E	. E
2. Average release rate for period	μCi/sec	. E	. E
3. Percent of Technical specification limit	%	. E	. E
C. Particulates			
1. Particulates with half-lives 8 days	Ci	. E	. E
2. Average release rate for period	μCi/sec	. E	. E
3. Percent of Technical specification limit	%	. E	. E
4. Gross alpha radioactivity	Ci	. E	. E
D. Tritium			
1. Total release	Ci	. E	. E
2. Average release rate for period	μCi/sec	. E	. E

TABLE 5.6.1-B

GASEOUS EFFLUENTS

Nuclides Released	Unit	CONTINUOUS MODE Quarter	CONTINUOUS MODE Quarter	BATCH MODE Quarter	BATCH MODE Quarter
1. Fission gases					
krypton-85	Ci	. E	. E	. E	. E
krypton-85m	Ci	. E	. E	. E	. E
krypton-87	Ci	. E	. E	. E	. E
krypton-88	Ci	. E	. E	. E	. E
xenon-133	Ci	. E	. E	. E	. E
xenon-135	Ci	. E	. E	. E	. E
xenon-135m	Ci	. E	. E	. E	. E
xenon-138	Ci	. E	. E	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
unidentified	Ci	. E	. E	. E	. E
Total for period	Ci	. E	. E	. E	. E
2. Iodines					
iodine-131	Ci	. E	. E	. E	. E
iodine-133	Ci	. E	. E	. E	. E
iodine-135	Ci	. E	. E	. E	. E
Total for period	Ci	. E	. E	. E	. E
3. Particulates					
strontium-89	Ci	. E	. E	. E	. E
strontium-90	Ci	. E	. E	. E	. E
cesium-134	Ci	. E	. E	. E	. E
cesium-137	Ci	. E	. E	. E	. E
barium-lanthanum-140	Ci	. E	. E	. E	. E
Others (specify)	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
unidentified	Ci	. E	. E	. E	. E

TABLE 5.6.1-C

LIQUID EFFLUENTS-SUMMATION OF ALL RELEASES

	Unit	Quarter	Quarter
A. Fission and activation products			
1. Total release (not including tritium, gases, alpha)	Ci	. E	. E
2. Average diluted concentration during period	μCi/ml	. E	. E
3. Percent of applicable limit	%	. E	. E
B. Tritium			
1. Total release	Ci	. E	. E
2. Average diluted concentration during period	μCi/ml	. E	. E
C. Dissolved and entrained gases			
1. Total release	Ci	. E	. E
2. Average diluted concentration during period	μCi/ml	. E	. E
3. Percent of applicable limit	%	. E	. E
D. Gross alpha radioactivity			
1. Total release	Ci	. E	. E
E. Volume of waste released (prior to dilution)	liters	. E	. E
F. Volume of dilution water used during period	liters	. E	. E

TABLE 5.6.1-D

LIQUID EFFLUENTS

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter	Quarter	Quarter	Quarter
strontium-89	Ci	. E	. E	. E	. E
strontium-90	Ci	. E	. E	. E	. E
cesium-134	Ci	. E	. E	. E	. E
cesium-137	Ci	. E	. E	. E	. E
iodine-131	Ci	. E	. E	. E	. E
cobalt-58	Ci	. E	. E	. E	. E
cobalt-60	Ci	. E	. E	. E	. E
iron-59	Ci	. E	. E	. E	. E
zinc-59	Ci	. E	. E	. E	. E
manganese-54	Ci	. E	. E	. E	. E
chromium-51	Ci	. E	. E	. E	. E
zirconium-niobium-95	Ci	. E	. E	. E	. E
molybdenum-99	Ci	. E	. E	. E	. E
technetium-99m	Ci	. E	. E	. E	. E
barium-lanthanum-140	Ci	. E	. E	. E	. E
cerium-141	Ci	. E	. E	. E	. E
Other (specify)	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
	Ci	. E	. E	. E	. E
unidentified	Ci	. E	. E	. E	. E
Total for period (above)	Ci	. E	. E	. E	. E
xenon-133	Ci	. E	. E	. E	. E
xenon-135	Ci	. E	. E	. E	. E

TABLE 5.6.1-E

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A. SOLID WASTE SHIPPED OFFSITE FOR BURIAL OR DISPOSAL (Not irradiated fuel)

1. Type of waste	Unit	6-month Period
a. Spent resins, filter sludges, evaporator bottoms, etc.	m^3 Ci	E E
b. Dry compressible waste, contaminated equip etc.	m^3 Ci	E E
c. Irradiated components, control rods, etc.	m^3 Ci	E E
d. Other (describe)	m^3 Ci	E E

2. SOLID WASTE DISPOSITION

Number of Shipments	Mode of Transportation	Destination

B. IRRADIATED FUEL SHIPMENTS (Disposition)

Number of Shipments	Mode of Transportation	Destination

1. Prompt Reports

Those events requiring prompt reports shall be reported within 24 hours by telephone, telegraph, or facsimile transmission to the Director of the Regional Office of Inspection and Enforcement and within 10 days by a written report to the Director of the Office of Inspection and Enforcement.

2. 30-Day Reports

Those events not requiring prompt reports shall be reported within 30 days by a written report to the Director of the Regional Office of Inspection and Enforcement with a copy to the Director of the Office of Inspection and Enforcement.

The reporting schedule for reports concerning limiting conditions shall be reported on the 30-day schedule. Reports concerning unusual or important events shall be reported on the prompt schedule.

Written 10-day and 30-day reports and to the extent possible the preliminary telephone, telegraph, or facsimile reports shall: a) describe, analyze, and evaluate the occurrence, including extent and magnitude of the impact, b) describe the cause of the occurrence and c) indicate the corrective action (including any significant changes made in procedures) taken to preclude repetition of the occurrence and to prevent similar occurrences involving similar components or systems.

The significance of an unusual or apparently important event with regard to environmental impact may not be obvious or fully appreciated at the time of occurrence. In such cases, the NRC shall be informed promptly of changes in the assessment of the significance of the event and a corrected report shall be submitted as expeditiously as possible.

5.6.2.b <u>Radioactive Effluent Reports</u>

<u>Liquid Radioactive Wastes Report</u>

If the cummulative releases of radioactive materials in liquid effluents, excluding tritium and dissolved gases, should exceed one-half the design objective annual quantity during any calendar quarter, the licensee shall make an investigation to identify the causes of such releases and define and initiate a program of action to reduce such releases to the design objective levels. A written report of these actions shall be submitted to the NRC within 30 days from the end of the quarter during which the release occurred.

<u>Gaseous Radioactive Wastes Report</u>

Should the conditions a), b), or c) listed below exist, the licensee shall make an investigation to identify the causes of the release rates and define and

and initiate a program of action to reduce the release rates to design objective levels. A written report of these actions shall be submitted to the NRC within 30 days from the end of the quarter during which the releases occurred.

a. If the average release rate of noble gases for the site during any calendar quarter exceeds one-half the design objective annual quantity.

b. If the average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter exceeds one-half the design objective annual quantity.

c. If the amount of iodine-131 released during any calendar quarter is greater than 0.5 Ci/reactor.

Unplanned or Uncontrolled Release Report

Any unplanned or uncontrolled offsite release of radioactive materials in excess of 0.5 curie in liquid or in excess of 5 curies of noble gases or 0.02 curie of radioiodines in gaseous form requires notification. This notification must be made by a written report within 30 days to the NRC. The report shall describe the event, identify the causes of the unplanned or uncontrolled release and report actions taken to prevent recurrence.

5.6.2.c Radiological Environmental Surveillance Reports

If a confirmed measured level of radioactivity in an environmental medium exceeds ten times the control station value, a written report shall be submitted to the Director of the NRC Regional Office (with a copy to the Director, Office of Nuclear Reactor Regulation) within 10 days after confirmation of the validity of the measured level. Confirmation shall be completed at the earliest time consistent with the analysis, but in any case, within 30 days. This report shall include an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result.

5.6.3 Changes in Environmental Technical Specifications

Request for changes in environmental technical specifications shall be submitted to the Director of Nuclear Reactor Regulation for review and authorization. The request shall include an evaluation of the environmental impact of the proposed change.

5.7 Records Retention

5.7.1 Records and logs relative to the following areas shall be made and retained for the life of the plant:

a. Records and drawings detailing plant design changes and modifications made to systems and equipment as described in 5.3.3.F.S.

b. Records of all environmental surveillance data.

c. Records to demonstrate compliance with the limiting conditions in Section 2.

5.7.2 All other records and logs relating to the environmental technical specifications shall be retained for five years following logging or recording. These shall include (but are not limited to) the following:

a. Details or any abnormal operating conditions having an effect on the environment, and actions taken to correct those conditions.

b. Maintenance activities to environment monitoring equipment, including but not limited to:

1) routine maintenance and component replacement,

2) equipment failures,

3) replacement of principal items of equipment.

c. Records of radioactivity levels in liquid and gaseous wastes released to the environment.

d. All reviews, including actions taken and reasons therefor, required in Sections 2, 3, and 4 of this specification.

[illegible] [illegible]

6.0 SPECIAL CONDITIONS

6.1 Light Screen to Minimize Turtle Disorientation

Australian pine or other suitable plants (i.e., native vegetation such as live oak, native figs, wild tamarind, and others) shall be planted and maintained as a light screen, along the beach dune line bordering the plant property to minimize turtle disorientation.