CRYSTAL RIVER
NUCLEAR GENERATING PLANT
UNIT 3
TECHNICAL SPECIFICATIONS

APPENDIX "A"
TO
LICENSE NO. DPR - 72

50-302

DECEMBER 3, 1976 Rec'd w/ltr 12-3-76

ISSUED BY THE UNITED STATES NUCLEAR REGULATORY COMMISSION

Crystal River Unit 3 Nuclear Generating Plant

Technical Specifications

Appendix "A"

to

License No. DPR-72

INDEX

DEFINITIONS

INDEX

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

BASES

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

INDEX

BASES

BASES

SECTION | PAGE

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

SECTION | PAGE

INDEX

BASES

BASES

SECTION PAGE

INDEX

DESIGN FEATURES

INDEX

ADMINISTRATIVE CONTROLS

INDEX

ADMINISTRATIVE CONTROLS

SECTION 1.0

DEFINITIONS

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 2452 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, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment, that are required for the system, subsystem, train, component or device to perform its function(s), are also capable of performing their related support function(s).

REPORTABLE OCCURRENCE

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

CONTAINMENT INTEGRITY

1.8 CONTAINMENT INTEGRITY shall exist when:

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

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

 2. Closed by manual valves, blind flanges, or deactivated automatic valves secured in their closed positions, except as provided in Table 3.6-1 of Specification 3.6.3.1.

b. All equipment hatches are closed and sealed,

c. Each airlock is OPERABLE pursuant to Specification 3.6.1.3,

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

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

CHANNEL CALIBRATION

1.9 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with 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. CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

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

CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be:

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

b. Bistable channels - the injection of a simulated signal into the channel sensor 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 ALTERATIONS 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 subcritical or would be subcritical from its present condition assuming:

a. No change in axial power shaping rod position, and

b. All control rod assemblies (safety and regulating) are fully inserted except for the single rod assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

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

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

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

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 that seal water flow from the reactor coolant pump seals.

QUADRANT POWER TILT

1.18 QUADRANT POWER TILT is defined by the following equation and is expressed in percent.

QUADRANT POWER TILT =

$$100 \left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1 \right)$$

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 thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites."

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

1.20 $\bar{E}$-AVERAGE DISINTEGRATION ENERGY 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 designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or designated components 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 POWER IMBALANCE

1.23 AXIAL POWER IMBALANCE shall be the THERMAL POWER in the top half of the core expressed as a percentage of RATED THERMAL POWER minus the THERMAL POWER in the bottom half of the core expressed as a percentage of RATED THERMAL POWER.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.24 The REACTOR PROTECTION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until power interruption at the control rod drive breakers.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.25 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

1.26 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 13 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	$\geq$ 0.99	> 5%	$\geq$ 280°F
2. STARTUP	$\geq$ 0.99	$\leq$ 5%	$\geq$ 280°F
3. HOT STANDBY	< 0.99	0	$\geq$ 280°F
4. HOT SHUTDOWN	< 0.99	0	280°F > T_{avg} > 200°F
5. COLD SHUTDOWN	< 0.99	0	$\leq$ 200°F
6. REFUELING**	$\leq$ 0.95	0	$\leq$ 140°F

*Excluding decay heat.

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

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 the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the safety limit shown in Figure 2.1-2 for the various combinations of two, three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate safety limit, be in HOT STANDBY within one hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

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

ACTION:

MODES 1 and 2 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

MODES 3, 4 and 5 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

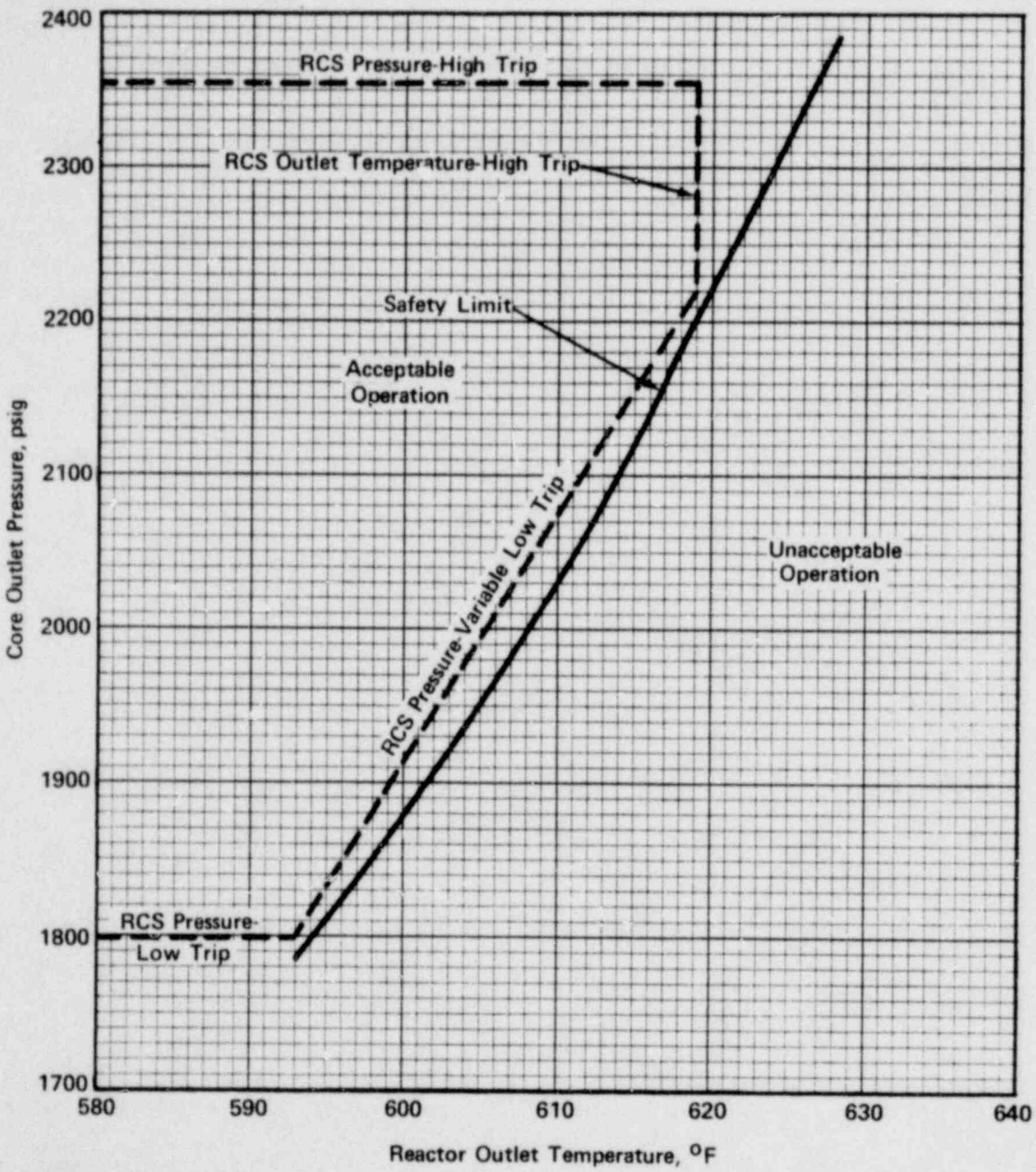
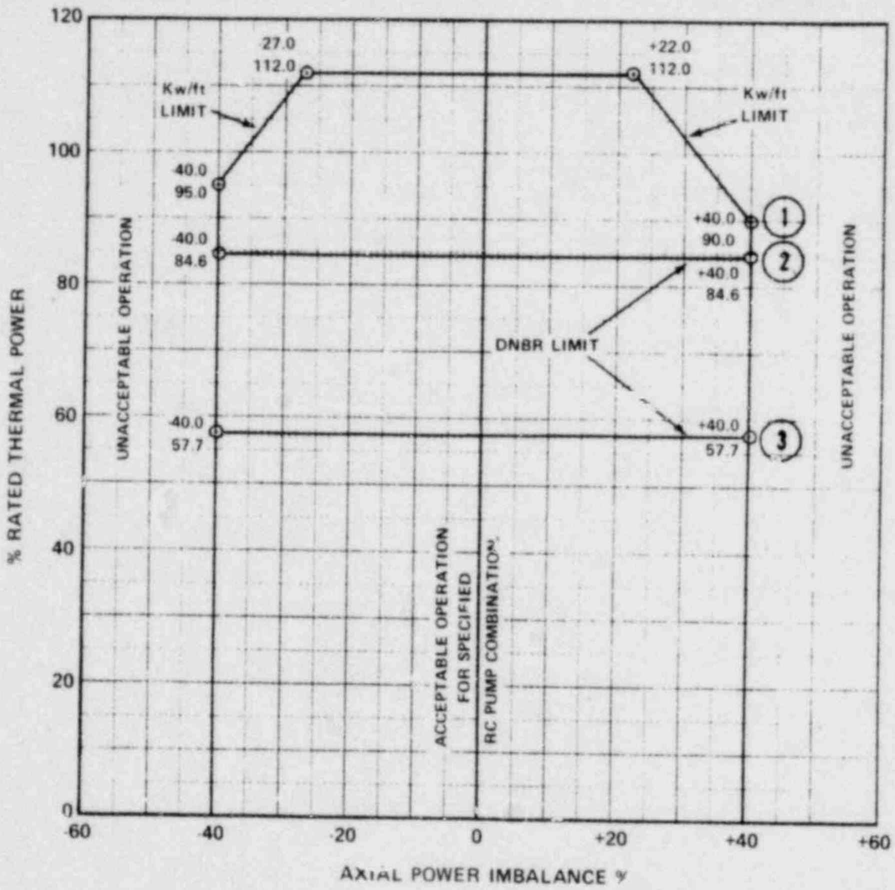


Figure 2.1-1
Reactor Core Safety Limit



CURVE	REACTOR COOLANT FLOW (lb/hr)
1	131.3×10^6
2	98.1×10^6
3	64.4×10^6

Figure 2.1-2

Reactor Core Safety Limit

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System 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 Protection System 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 PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating
	$\leq$ 78% of RATED THERMAL POWER with three pumps operating	$\leq$ 78% of RATED THERMAL POWER with three pumps operating
	$\leq$ 51.2% of RATED THERMAL POWER with one pump operating in each loop	$\leq$ 51.2% of RATED THERMAL POWER with one pump operating in each loop
3. RCS Outlet Temperature-High	$\leq$ 619°F	$\leq$ 619°F
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBLANCE(1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-1.
5. RCS Pressure-Low(1)	$\geq$ 1800 psig	$\geq$ 1800 psig
6. RCS Pressure-High	$\leq$ 2355 psig	$\leq$ 2355 psig
7. RCS Pressure-Variable Low(1)	$\geq$ (16.25 T_{out} °F - 7838) psig	$\geq$ (16.25 T_{out} °F - 7838) psig

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTION UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Reactor Containment Vessel Pressure High	$\leq$ 4 psig	$\leq$ 4 psig

(1) Trip may be manually bypassed when RCS pressure $\leq$ 1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of $\leq$ 1720 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

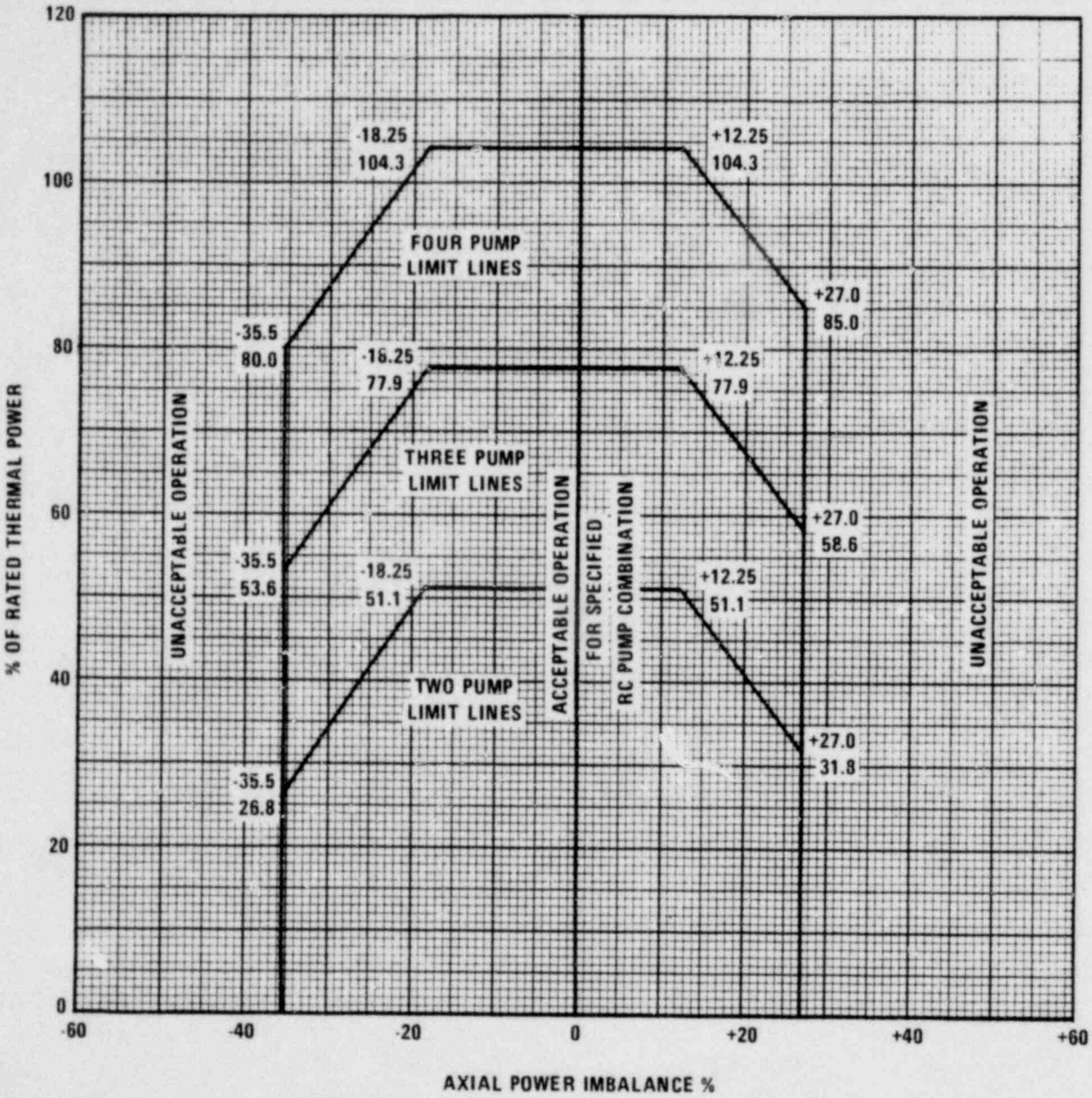


Figure 2.2-1

Trip Setpoint For Nuclear Overpower Based On RCS Flow and AXIAL POWER IMBALANCE

BASES

FOR

SAFETY LIMITS

AND

LIMITING SAFETY SYSTEM SETTINGS

NOTE

The summary statements contained in this section provide the bases for the specifications of Section 2.0 and are not considered a part of these technical specifications as provided in 10 CFR 50.36.

2.1 SAFETY LIMITS

BASES

2.1.1 and 2.1.2 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 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 would 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 DNB correlation. The 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 94.3 percent probability at a 99 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a minimum DNBR of 1.30 is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 131.3×10^6 lbs/hr, which is the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

$$F_Q^N = 2.67; \quad F_{\Delta H}^N = 1.78; \quad F_Z^N = 1.50$$

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod withdrawal, and form the core DNBR design basis.

BASES

The reactor trip envelope appears to approach the safety limit more closely than it actually does because the reactor trip pressures are measured at a location where the indicated pressure is about 30 psi less than core outlet pressure, providing a more conservative margin to the safety limit.

The curves of Figure 2.1-2 are based on the more restrictive of two thermal limits and account for the effects of potential fuel densification and potential fuel rod bow:

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of $F_Q^N = 2.67$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 19.7 kw/ft.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The specified flow rates for curves 1, 2, and 3 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop, respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.30 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR is equal to 15%, whichever condition is more restrictive.

These curves include the potential effects of fuel rod bow and fuel densification.

Using a local quality limit of 15% at the point of minimum DNBR as a basis for curve 3 of BASES Figure 2.1 is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the W-3 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of 15% is justified on the basis of experimental data.

For each curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 or a local quality at the point of minimum DNBR less than 15% for that particular reactor coolant pump situation. The 1.30 DNBR curve for four pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the four pump curve will be above and to the left of the other curves.

2.1.3 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 Boiler and Pressure Vessel Code which permits a maximum transient pressure of 110%, 2750 psig, of design pressure. The Reactor Coolant System piping, valves and fittings, are designed to USAS B 31.7, February, 1968 Draft Edition, which permits a maximum transient pressure of 110%, 2750 psig, of component design pressure. The Safety Limit of 2750 psig is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psig, 125% of design pressure, to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

The Shutdown Bypass provides for bypassing certain functions of the Reactor Protection System in order to permit control rod drive tests, zero power PHYSICS TESTS and certain startup and shutdown procedures. The purpose of the Shutdown Bypass RCS Pressure-High trip is to prevent normal operation with Shutdown Bypass activated. This high pressure trip setpoint is lower than the normal low pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The Nuclear Overpower Trip Setpoint of $\leq$ 5.0% prevents any significant reactor power from being produced. Sufficient natural circulation would be available to remove 5.0% of RATED THERMAL POWER if none of the reactor coolant pumps were operating.

Manual Reactor Trip

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

Nuclear Overpower

A Nuclear Overpower trip at high power level (neutron flux) provides reactor core protection against reactivity excursions which are too rapid to be protected by temperature and pressure protective circuitry.

During normal station operation, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, 114% was used in the safety analysis.

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip ≤ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

The power level trip setpoint produced by the reactor coolant system flow is based on a flux-to-flow ratio which has been established to accomodate flow decreasing transients from high power.

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate. Typical power level and low flow rate combinations for the pump situations of Table 2.2-1 are as follows:

1. Trip would occur when four reactor coolant pumps are operating if power is ≥ 104.5% and reactor flow rate is 100%, or flow rate is ≤ 95.7% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is ≥ 80% and reactor flow rate is 76.6%, or flow rate is ≤ 71.8% and power is 75%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is ≥ 51.0% and reactor flow rate is 48.8% or flow rate is ≤ 47.8% and the power level is 50.0%.

For safety calculations the maximum calibration and instrumentation errors for the power level were used.

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.044% for a 1% flow reduction.

RCS Pressure - Low, High and Variable Low

The High and Low trips are provided to limit the pressure range in which reactor operation is permitted.

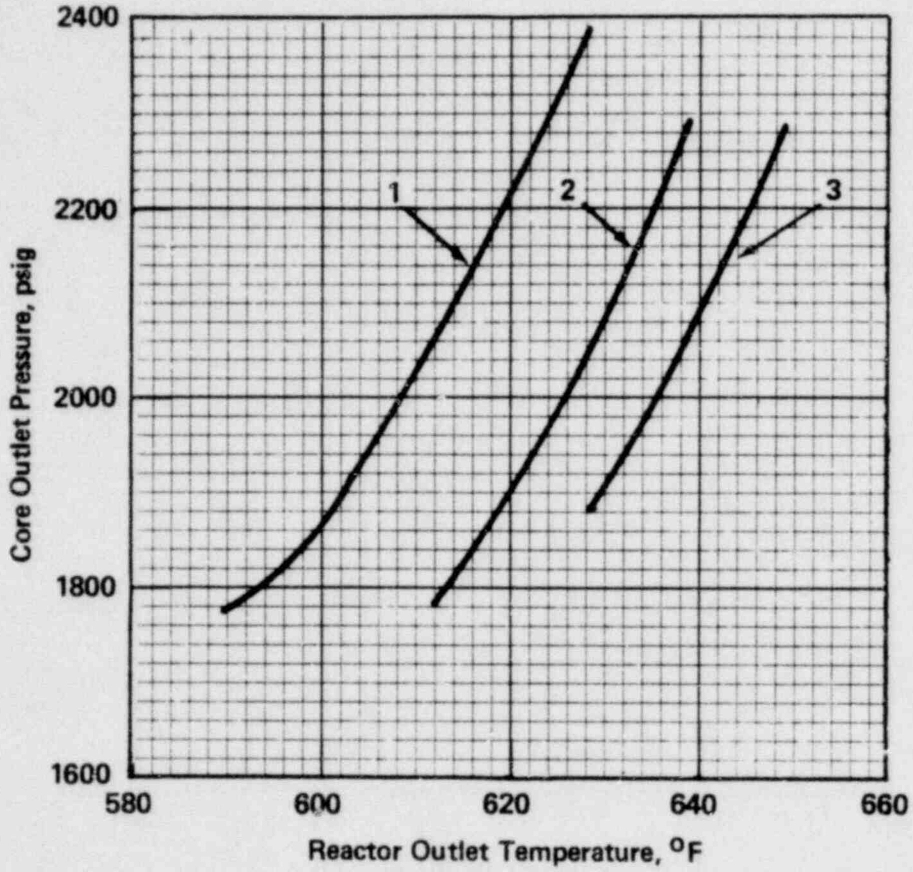
During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2355 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection, and is therefore set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1800 psig, and RCS Pressure-Variable Low, ($16.25\ T_{out}$°F-7838) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of ($16.25\ T_{out}$°F-7878) psig.

Reactor Containment Vessel Pressure - High

The Reactor Containment Vessel Pressure-High Trip Setpoint $\leq$ 4 psig, provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the containment vessel or a loss-of-coolant accident, even in the absence of a RCS Pressure -Low trip.



Reactor Coolant Flow

Curve	(LBS/HR)	Power	Pumps Operating (Type of Limit)
1	131.3×10^6 (100%)	112%	Four Pumps (DNBR Limit)
2	98.1×10^6 (74.7%)	84%	Three Pumps (DNBR Limit)
3	64.4×10^6 (49.0%)	57%	One Pump in each loop(Quality Limit)

Pressure/Temperature Limits at Maximum
Allowable Power for Minimum DNBR

BASES Figure 2.1

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 surveillance interval, and

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

SURVEILLANCE REQUIREMENTS (Continued)

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. Surveillance Requirements do not have to be performed on inoperable equipment.

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

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

a. For the time period from issuance of the Facility Operating License to the start of facility commercial operation, inservice testing of ASME Code Class 1, 2 and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition, and Addenda through Summer 1975, except where specific written relief has been granted by the Commission.

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

Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements. The provisions of Specification 4.0.2 are not applicable to surveillance intervals associated with inservice inspection and testing activities required by Section XI of the above ASME Boiler and Pressure Vessel Code and applicable Addenda.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 1% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 12,250 ppm boron or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1% $\Delta k/k$:

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

b. When in MODES 1 or 2#, at least once per 12 hours, by verifying that regulating rod groups withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2## within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading by consideration of the factors of e. below, with the regulating rod groups at the maximum insertion limit of Specification 3.1.3.6.

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

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

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3, 4 or 5, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. Control rod 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% $\Delta k/k$ 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.

BORON DILUTION

LIMITING CONDITION FOR OPERATION

3.1.1.2 The flow rate of reactor coolant through the Reactor Coolant System shall be ≥ 2700 gpm whenever a reduction in Reactor Coolant System boron concentration is being made.

APPLICABILITY: All MODES.

ACTION:

With the flow rate of reactor coolant through the Reactor Coolant System < 2700 gpm, immediately suspend all operations involving a reduction in boron concentration of the Reactor Coolant System.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The flow rate of reactor coolant through the Reactor Coolant System shall be determined to be ≥ 2700 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 DHR pump is in operation and supplying ≥ 2700 gpm through the Reactor Coolant System.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.9×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is < 95% of RATED THERMAL POWER,

b. Less positive than 0.0×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\geq$ 95% of RATED THERMAL POWER, and

c. Less negative than -3.0×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

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

b. At any THERMAL POWER, within 7 days after reaching a RATED THERMAL POWER equilbrium boron concentration of 300 ppm.

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

#See Special Test Exception 3.10.2.

MINIMUM TEMPERATURE FOR CRITICALITY

LIMITING CONDITION FOR OPERATION

3.1.1.4 The Reactor Coolant System lowest loop temperature (T_{avg}) shall be $\geq$ 525°F.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With a Reactor Coolant System loop temperature (T_{avg}) < 525°F, restore T_{avg} to within its limit within 15 minutes or be in HOT STANDBY within the next 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.1.4 The RCS temperature (T_{avg}) shall be determined to be $\geq$ 525°F:

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

b. At least once per 30 minutes when the reactor is critical and the Reactor Coolant System T_{avg} is less than 530°F.

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

3/4.1.2 BORATION SYSTEMS

FLOW PATHS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 At least one of the following boron injection flow paths shall be OPERABLE.

a. A flow path from the concentrated boric acid storage system via a boric acid pump and a makeup or decay heat removal (DHR) pump to the Reactor Coolant System, if only the boric acid storage system in Specification 3.1.2.8a is OPERABLE, or

b. A flow path from the borated water storage tank via a makeup or DHR pump to the Reactor Coolant System if only the borated water storage tank in Specification 3.1.2.8b 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 verifying that the pipe temperature of the heat traced portion of the flow path is $\geq$ 105°F when a flow path from the concentrated boric acid storage system is used, and

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 Each of the following boron injection flow paths shall be OPERABLE:

a. A flow path from the concentrated boric acid storage system via a boric acid pump and makeup or decay heat removal (DHR) pump to the Reactor Coolant System, and

b. A flow path from the borated water storage tank via makeup or DHR pump to the Reactor Coolant System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the flow path from the concentrated boric acid storage system inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVIELLANCE REQUIREMENTS

4.1.2.2 Each of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the pipe temperature of the heat traced portion of the flow path from the concentrated boric acid storage system is $\geq$ 105°F.

SURVEILLANCE REQUIREMENTS (Continued)

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.

REACTIVITY CONTROL SYSTEMS

MAKEUP PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one makeup pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODE 5*.

ACTION:

With no makeup pump CPERABLE, suspend all operations involving positive reactivity changes until at least one makeup pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

*RCS Pressure $\geq$ 150 psig.

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4 At least two makeup pumps shall be OPERABLE.

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

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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*With RCS pressure $\geq$ 150 psig.

DECAY HEAT REMOVAL PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.5 At least one decay heat removal (DHR) pump in the boron injection flow path required by Specification 3.1.2.1 or 3.1.2.2 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODES 4*, 5* and 6.

ACTION:

With no DHR pump OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes until at least one DHR pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

*RCS Pressure < 150 psig.

REACTIVITY CONTROL SYSTEMS

BORIC ACID PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.7 At least one boric acid pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least one boric acid pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

a. A concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 5000 gallons,
2. Between 12,250 and 14,000 ppm of boron, and
3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A minimum contained borated water volume of 13,500 gallons,
2. A minimum boron concentration of 2270 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 CORE ALTERATION or positive reactivity changes until at least one borated water source is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

a. At least once per 7 days by:

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

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying the concentrated boric acid storage system solution temperature when it is the source of borated water.

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

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

a. The concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 5000 gallons,

2. Between 12,500 and 14,000 ppm of boron, and

3. A minimum solution temperature of 105°F.

b. The borated water storage tank (BWST) with:

1. A contained borated water volume of between 415,200 and 449,000 gallons,

2. Between 2270 and 2450 ppm of boron, and

3. A minimum solution temperature of 40°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore the concentrated boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

SURVEILLANCE REQUIREMENTS

4.1.2.9 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

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

3. Verifying the concentrated boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the BWST temperature when the outside air temperature is $< 40°F$.

REACTIVITY CONTROL SYSTEMS

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT - SAFETY AND REGULATING ROD GROUPS

LIMITING CONDITION FOR OPERATIONS

3.1.3.1 All control (safety and regulating) rods shall be OPERABLE and positioned within $\pm$ 6.5% (indicated position) of their group average height.

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

b. With more than one control rod inoperable or misaligned from its group average height by more than $\pm$ 6.5% (indicated position), be in at least HOT STANDBY within 6 hours.

c. With one control rod inoperable due to causes other than addressed in ACTION a, above, or misaligned from its group average height by more than $\pm$ 6.5% (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The control rod is restored to OPERABLE status within the above alignment requirements, or

   2. The control rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) An analysis of the potential ejected rod worth is performed within 72 hours and the rod worth is determined to be $\leq$ 1.0% Δk at zero power and $\leq$ 0.65% Δk at RATED THERMAL POWER for the remainder of the fuel cycle, and

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

*See Special Test Exceptions 3.10.1 and 3.10.2.

ACTION: (Continued)

c) A power distribution map is obtained from the incore detectors and F_Q and $F^N_{\Delta H}$ are verified to be within their limits within 72 hours, and

d) The THERMAL POWER level is reduced to ≤ 60% of the THERMAL POWER allowable for the reactor coolant pump combination within one hour and within the next 4 hours the Nuclear Overpower Trip Setpoint is reduced to ≤ 70% of the THERMAL POWER allowable for the reactor coolant pump combination, or

e) The remainder of the rods in the group with the inoperable rod are aligned to within ± 6.5% of the inoperable rod within one hour while maintaining the rod sequence, insertion and overlap limits of Figures 3.1-1, 3.1-2, 3.1-3, 3.1-4, 3.1-5 and 3.1-6; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.6 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each control rod shall be determined to be within the group average height limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Asymmetric Rod Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

4.1.3.1.2 Each control rod not fully inserted shall be determined to be OPERABLE by movement of at least 3% in any one direction at least once every 31 days.

GROUP HEIGHT - AXIAL POWER SHAPING ROD GROUP

LIMITING CONDITION FOR OPERATION

3.1.3.2 All axial power shaping rods (APSR) shall be OPERABLE, unless fully withdrawn, and shall be positioned within ± 6.5% (indicated position) of their group average height.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With a maximum of one APSR inoperable or misaligned from its group average height by more than ± 6.5% (indicated position), operation may continue provided that within 2 hours:

a. The APSR group is positioned such that the misaligned rod is restored to within limits for the group average height, or

b. It is determined that the imbalance limits of Specification 3.2.1 are satisfied and movement of the APSR group is prevented while the rod remains inoperable or misaligned.

SURVEILLANCE REQUIREMENTS

4.1.3.2.1 The position of each APSR rod shall be determined to be within the group average height limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Asymmetric Control Rod Monitor is inoperable, then verify the individual rod positions at least once per 4 hours.

4.1.3.2.2 Unless all APSR are fully withdrawn, each APSR shall be determined to be OPERABLE by moving the individual rod at least 3% at least once every 31 days.

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*See Special Test Exceptions 3.10.1 and 3.10.2.

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.3 All safety, regulating and axial power shaping control rod reed switch position indicator channels and pulse stepping position indicator channels shall be OPERABLE and capable of determining the control rod positions within $\pm$ 2%.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one reed switch position indicator channel per control rod group or one pulse stepping position indicator channel per control rod group inoperable either:

   1. Reduce THERMAL POWER to $\leq$ 60% of the THERMAL POWER allowable for the Reactor Coolant Pump combination and reduce the Nuclear Overpower Trip Setpoint to $\leq$ 70% of the THERMAL POWER allowable for the reactor coolant pump combination within 8 hours, or

   2. Operation may continue provided:

      a) The position of the control rod with the inoperable position indicator is verified within 8 hours by actuating its 0%, 25%, 50%, 75% or, 100% position reference indicator, and

      b) The control rod group(s) containing the inoperable position indicator channel is subsequently maintained at the 0%, 25%, 50%, 75% or, 100% withdrawn position and verified at this position at least once per 12 hours thereafter, and

      c) Operation is within the limits of Specification 3.1.3.6.

b. With more than one pulse stepping position indicator channel 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.

POSITION INDICATOR CHANNELS (Continued)

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each reed switch and pulse stepping position indicator channel shall be determined to be OPERABLE by verifying that the pulse stepping position indicator channels and the reed switch position indicator channels agree within 2% at least once per 12 hours except during time intervals when the Asymmetric Rod Monitor is inoperable, then compare the pulse stepping position indicator and reed switch position indicator channels at least once per 4 hours.

ROD DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual safety and regulating rod drop time from the fully withdrawn position shall be $\leq$ 1.66 seconds from power interruption at the control rod drive breakers to 3/4 insertion (25% position) with:

a. $T_{avg} \geq 525°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With the drop time of any safety or regulating rod determined to exceed the above limit, restore the rod drop time to within the above limit prior to proceeding to MODE 1 and 2.

b. With the rod drop times within limits but determined with less than 4 reactor coolant pumps operating, operation may proceed provided that THERMAL POWER is restricted to less than or equal to the THERMAL POWER allowable for the reactor coolant pump combination operating at the time of rod drop time measurement.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The rod drop time of safety and regulating rods shall be demonstrated through measurement prior to reactor criticality:

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

b. For specifically affected individual rods following any maintenance on or modification to the control rod drive system which could affect the drop time of those specific rods, and

c. At least once every 18 months.

SAFETY ROD INSERTION LIMIT

LIMITING CONDITION FOR OPERATION

3.1.3.5 All safety rods shall be fully withdrawn.

APPLICABILITY: 1* and 2*#.

ACTION:

With a maximum of one safety rod not fully withdrawn, except for surveillance testing pursuant to Specification 4.1.3.1.2, within one hour either:

a. Fully withdraw the rod or

b. Declare the rod to be inoperable and apply Specification 3.1.3.1.

SURVEILLANCE REQUIREMENTS

4.1.3.5 Each safety rod shall be determined to be fully withdrawn:

a. Within 15 minutes prior to withdrawal of any regulating rod during an approach to reactor criticality.

b. At least once per 12 hours thereafter.

*See Special Test Exception 3.10.1 and 3.10.2.
#With $K_{eff} \geq 1.0$.

REGULATING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.6 The regulating rod groups shall be limited in physical insertion as shown on Figures 3.1-1, 3.1-2, 3.1-3, 3.1-4, 3.1-5 and 3.1-6 with a rod group overlap of 25 $\pm$ 5% between sequential withdrawn groups 5 and 6, and 6 and 7.

APPLICABILITY: MODES 1* and 2*#.

ACTION:

With the regulating rod groups inserted beyond the above insertion limits, or with any group sequence or overlap outside the specified limits, except for surveillance testing pursuant to Specification 4.1.3.1.2, either:

a. Restore the regulating groups to within the limits within 2 hours, or

b. Reduce THERMAL POWER to less than or equal to that fraction of RATED THERMAL POWER which is allowed by the rod group position using the above figures within 2 hours, or

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

*See Special Test Exceptions 3.10.1 and 3.10.2.
#With $K_{eff} \geq 1.0$.

REGULATING ROD INSERTION LIMITS

SURVEILLANCE REQUIREMENTS

4.1.3.6 The position of each regulating group shall be determined to be within the insertion, sequence and overlap limits at least once every 12 hours except when:

a. The regulating rod insertion limit alarm is inoperable, then verify the groups to be within the insertion limits at least once per 4 hours;

b. The control rod drive sequence alarm is inoperable, then verify the groups to be within the sequence and overlap limits at least once per 4 hours.

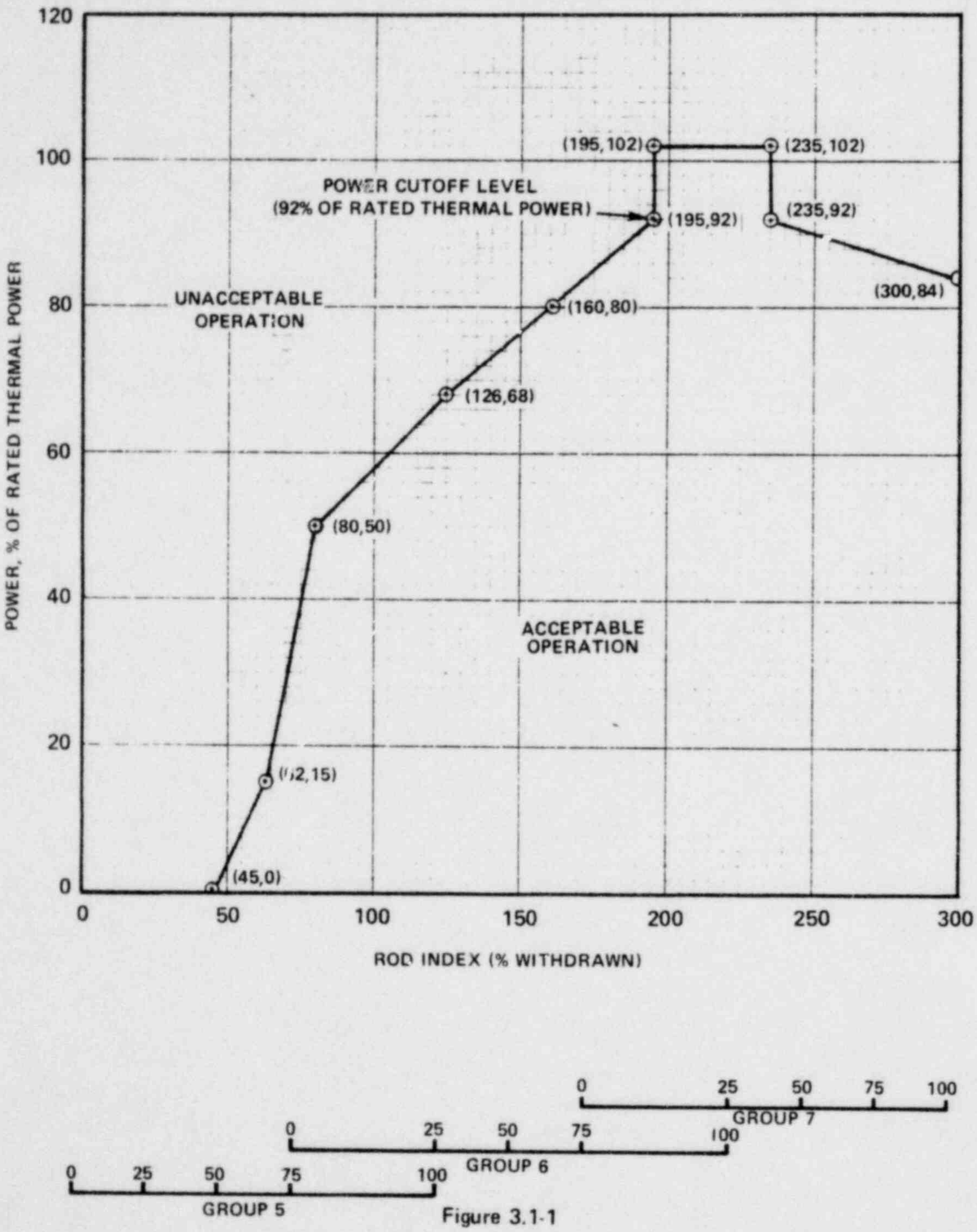


Figure 3.1-1

Regulating Rod Group Insertion Limits for 4 Pump Operation up to Control Rod Interchange 250 ± 10 EFPD

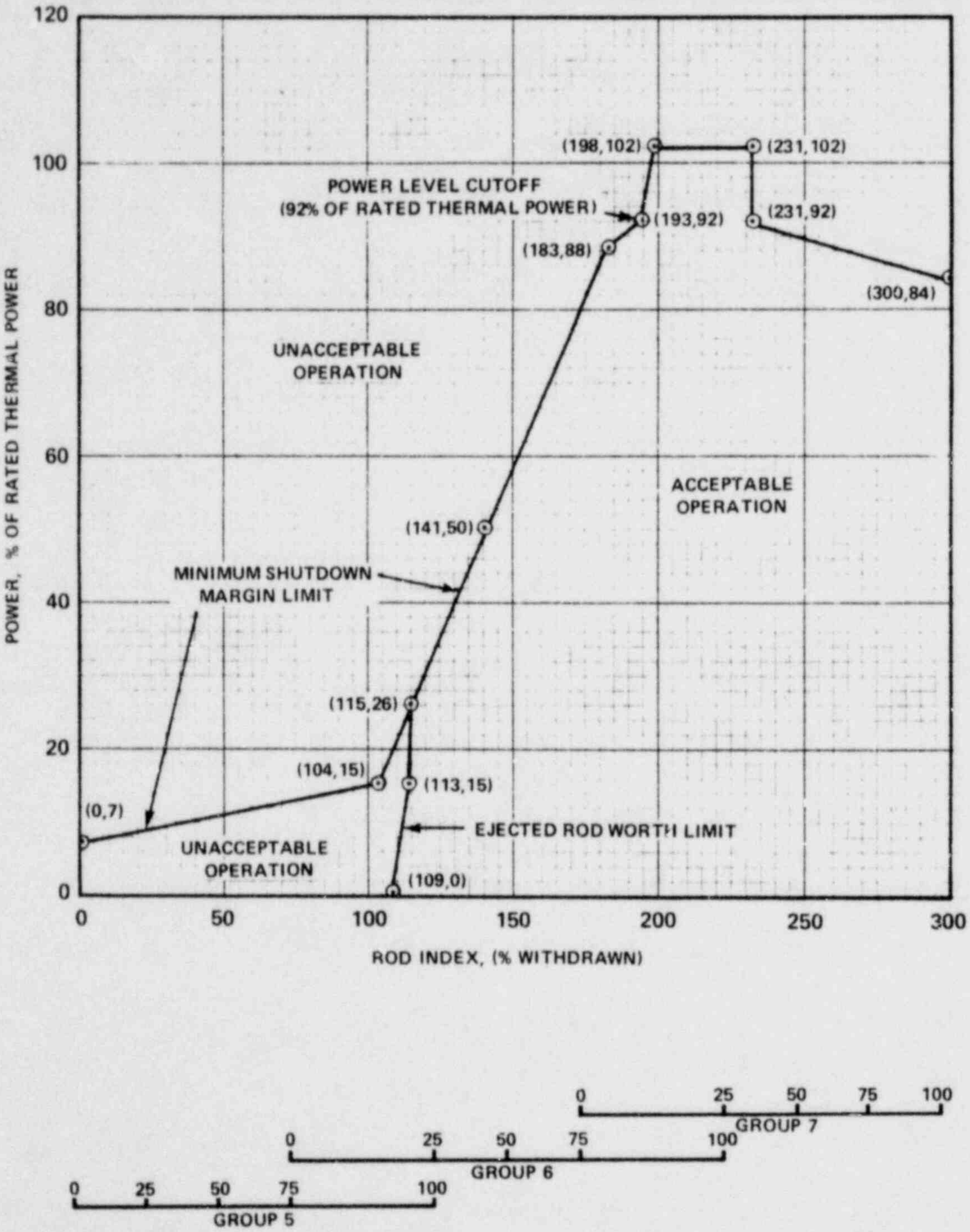


Figure 3.1-2 Regulating Rod Group Insertion Limits for 4 Pump Operation from Control Rod Interchange 250 $\pm$ 10 EFPD to 400 $\pm$ 10 EFPD

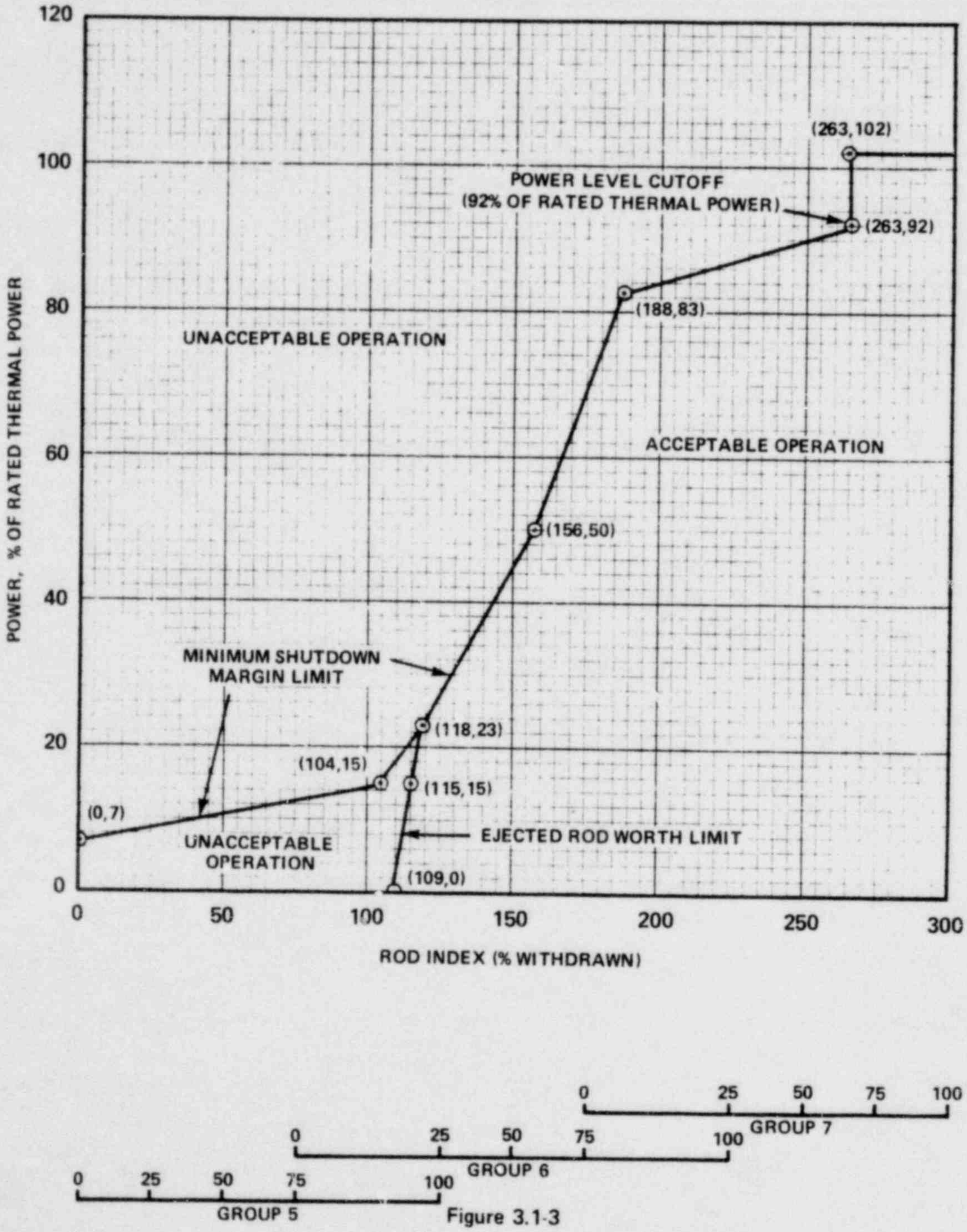


Figure 3.1-3

Regulating Rod Group Insertion Limits for 4 Pump Operation
After 400 ± 10 EFPD

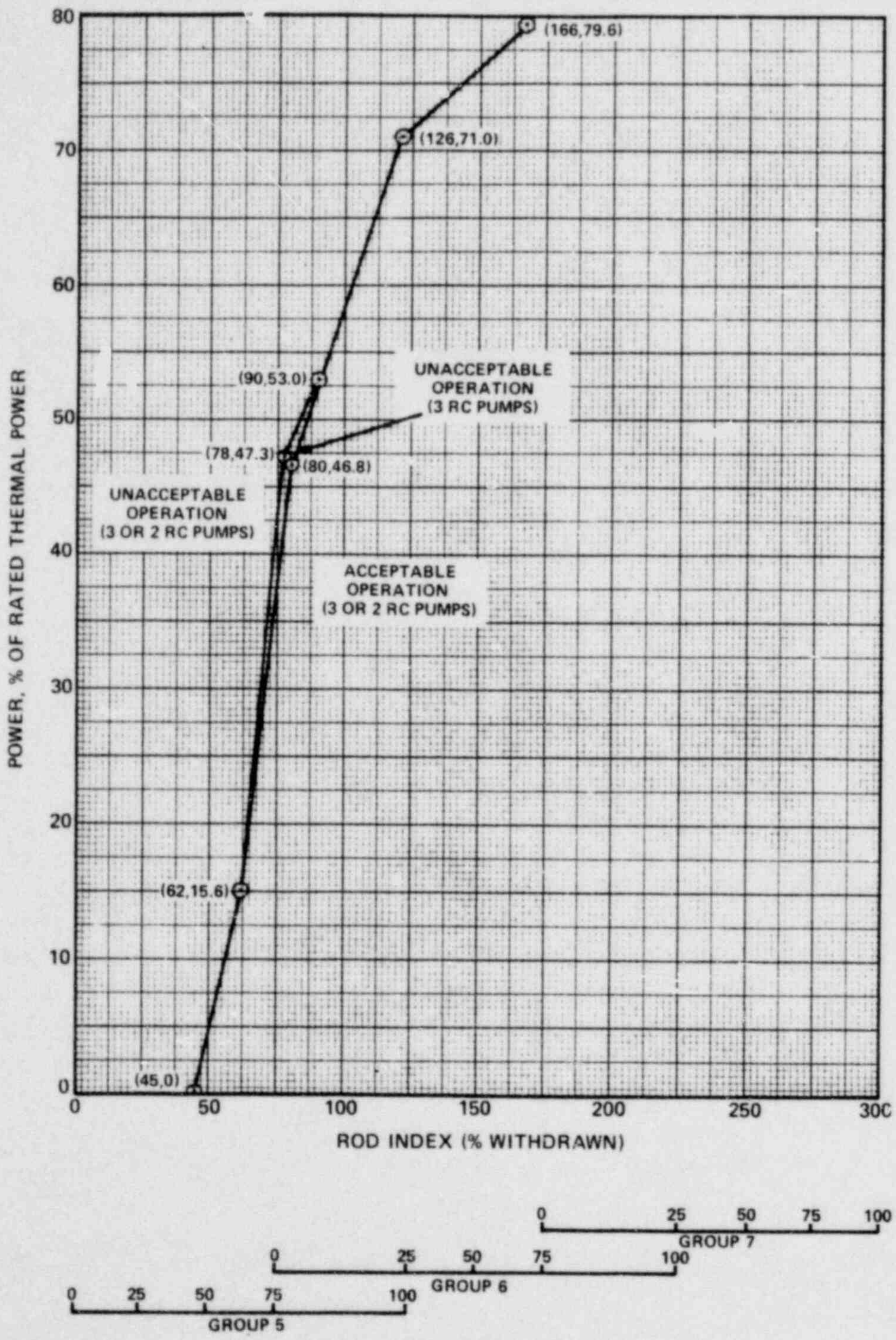


Figure 3.1-4

Regulating Rod Group Insertion Limits for 3 and 2 Pump Operation up to Control Rod Interchange 250 ± 10 EFPD

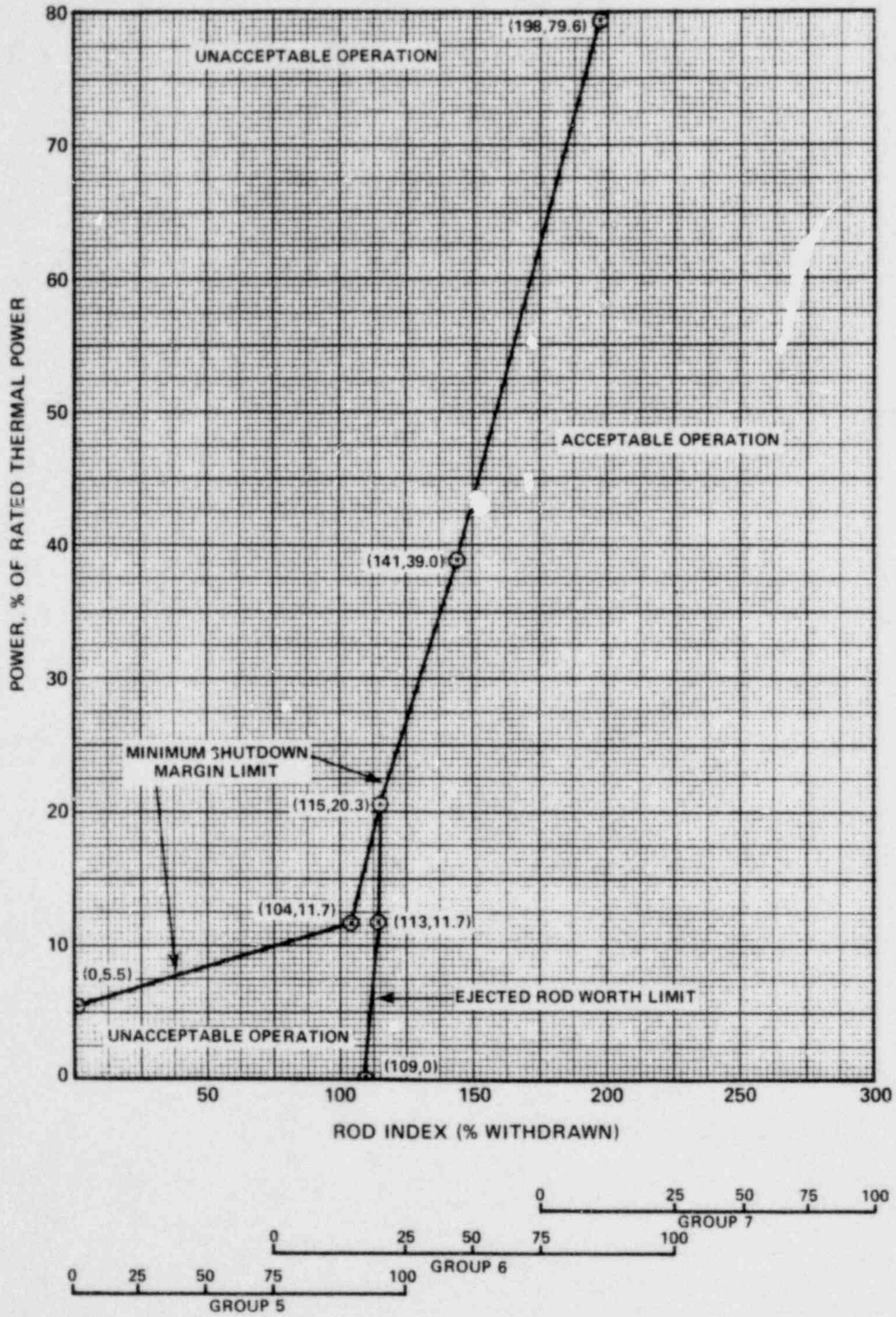


Figure 3.1-5

Regulating Rod Group Insertion Limits for 3 Pump Operation
After Control Rod Interchange 250 ± 10 EFPD

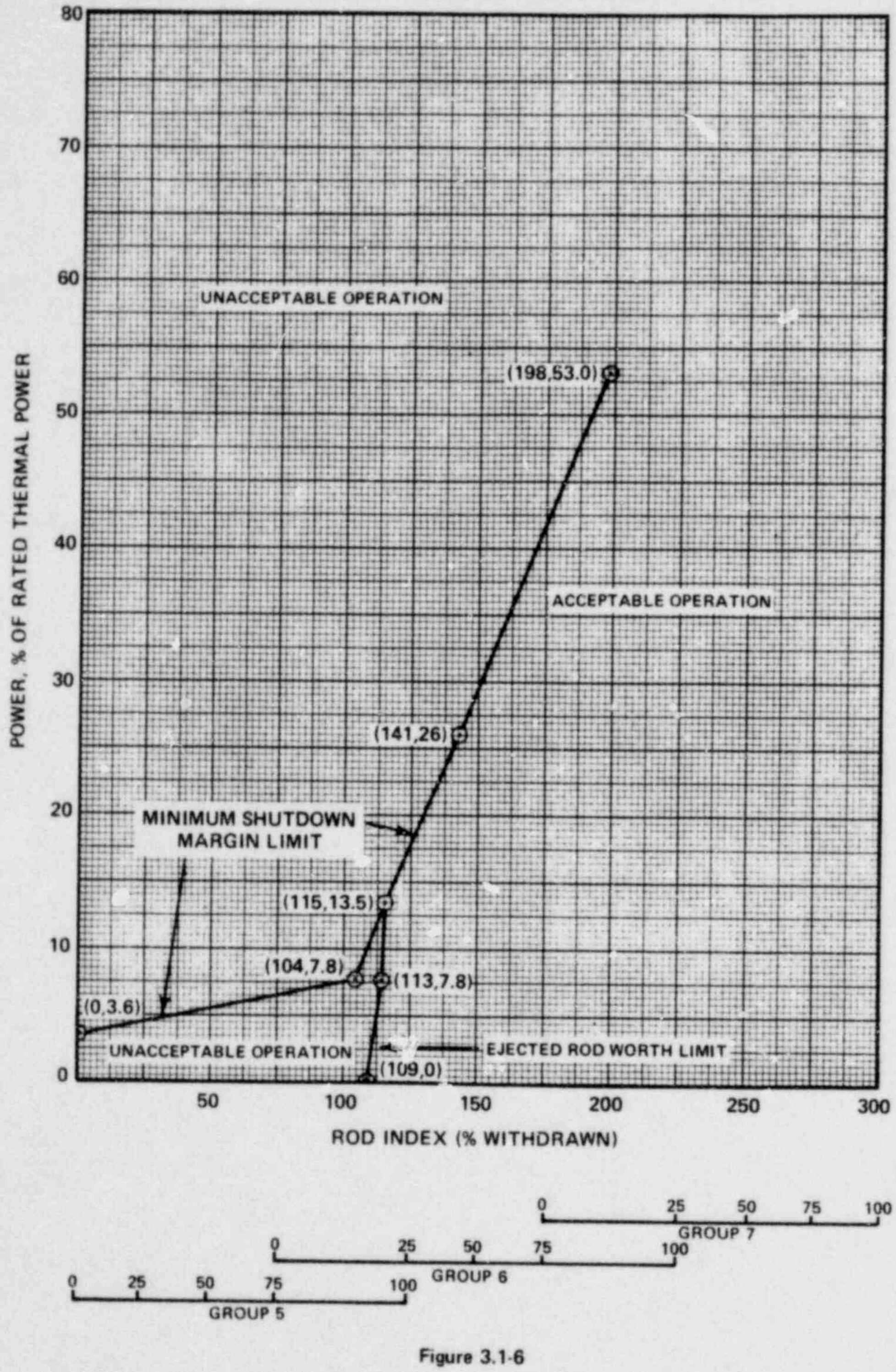


Figure 3.1-6

Regulating Rod Group Insertion Limits for 2 Pump Operation
After Control Rod Interchange 250 ± 10 EFPD

ROD PROGRAM

LIMITING CONDITION FOR OPERATION

3.1.3.7 Each control rod (safety, regulating and APSR) shall be programmed to operate in the core position and rod group specified in Figure 3.1-7 and 3.1-8.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With any control rod not programmed to operate as specified above, be in HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.3.7

a. Each control rod shall be demonstrated to be programmed to operate in the specified core position and rod group by:

   1. Selection and actuation from the control room and verification of movement of the proper rod as indicated by both the absolute and relative position indicators:

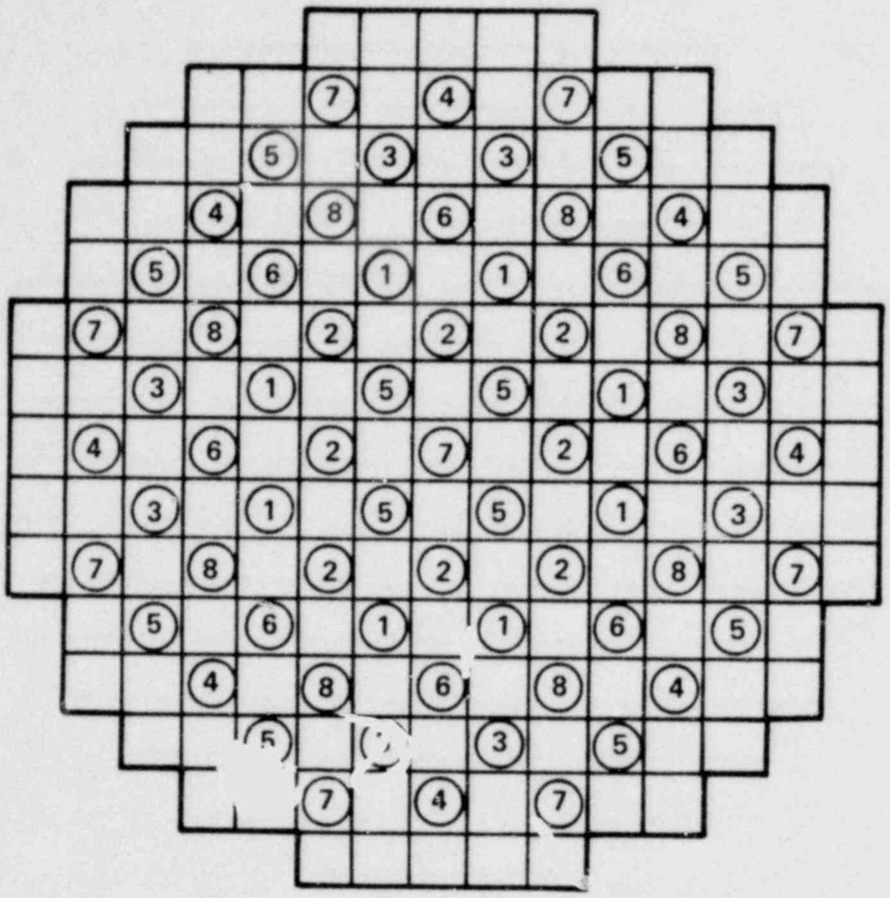
      a) For all control rods, after the control rod drive patchs are locked subsequent to test, reprogramming or maintenance within the panels.

      b) For specifically affected individual rods, following maintenance, test, reconnection or modification of power or instrumentation cables from the control rod drive control system to the control rod drive.

   2. Verifying that each cable that has been disconnected has been properly matched and reconnected to the specified control rod drive.

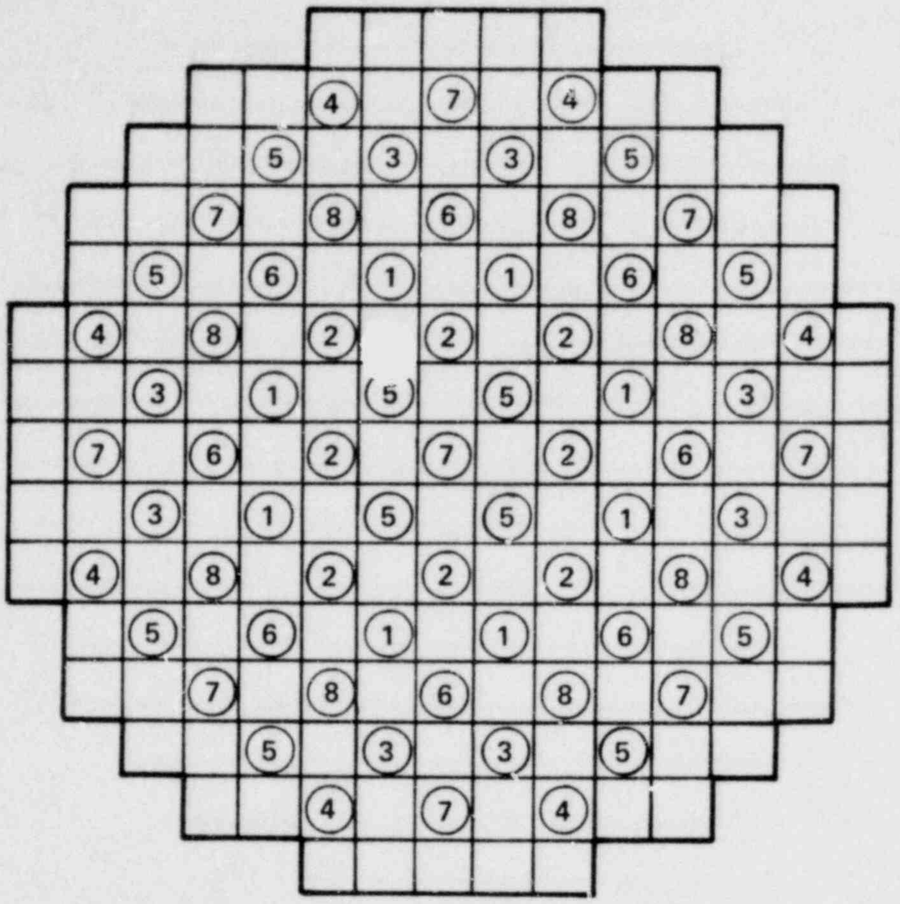
b. At least once each 7 days, verify that the control rod drive patch panels are locked.

*See Special Test Exceptions 3.10.1 and 3.10.2.



GROUP	NO. RODS	PURPOSE
1	8	SAFETY
2	8	SAFETY
3	8	SAFETY
4	8	SAFETY
5	12	DOPPLER
6	8	DOPPLER
7	9	TRANSIENT
8	8	APSR

Figure 3.1-7
Control Rod Core Location and Group Assignments up to 250 ± 10 EFPD



GROUP	NO. RODS	PURPOSE
1	8	SAFETY
2	8	SAFETY
3	8	SAFETY
4	8	SAFETY
5	12	DOPPLER
6	8	DOPPLER
7	9	TRANSIENT
8	8	APSR

Figure 3.1-8

Control Rod Core Location and Group Assignments after 250 ± 10 EFPD

XENON REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.3.8 THERMAL POWER shall not be increased above the power level cutoff specified in Figures 3.1-1, 3.1-2, and 3.1-3 unless xenon reactivity is within 10 percent of the equilibrium value for RATED THERMAL POWER and is approaching stability.

APPLICABILITY: MODE 1.

ACTION:

With the requirements of the above specification not satisfied, reduce THERMAL POWER to less than or equal to the power level cutoff within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.3.8 Xenon reactivity shall be determined to be within 10% of the equilibrium value for RATED THERMAL POWER and to be approaching stability prior to increasing THERMAL POWER above the power level cutoff.

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

3.2.1 AXIAL POWER IMBALANCE shall be maintained within the limits shown on Figures 3.2-1 and 3.2-2.

APPLICABILITY: MODE 1 above 40% of RATED THERMAL POWER.*

ACTION:

With AXIAL POWER IMBALANCE exceeding the limits specified above, either:

a. Restore the AXIAL POWER IMBALANCE to within its limits within 15 minutes, or

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

SURVEILLANCE REQUIREMENTS

4.2.1 The AXIAL POWER IMBALANCE shall be determined to be within limits in each core quadrant at least once every 12 hours when above 40% of RATED THERMAL POWER except when an AXIAL POWER IMBALANCE monitor is inoperable, then calculate the AXIAL POWER IMBALANCE in each core quadrant with an inoperable monitor at least once per hour.

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*See Special Test Exception 3.10.1.

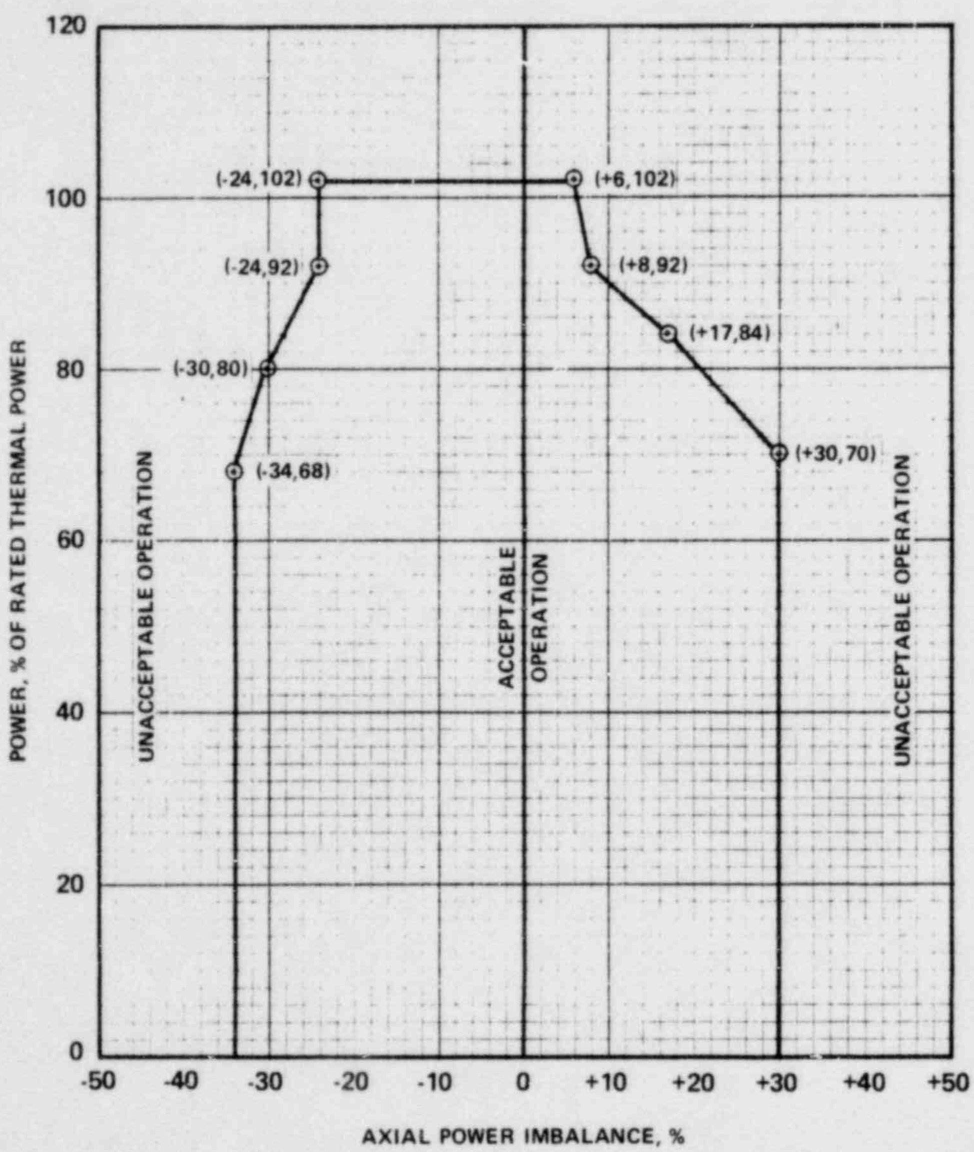


Figure 3.2-1
AXIAL POWER IMBALANCE Envelope for Operation
Up to 400 $\pm$ 10 EFPD

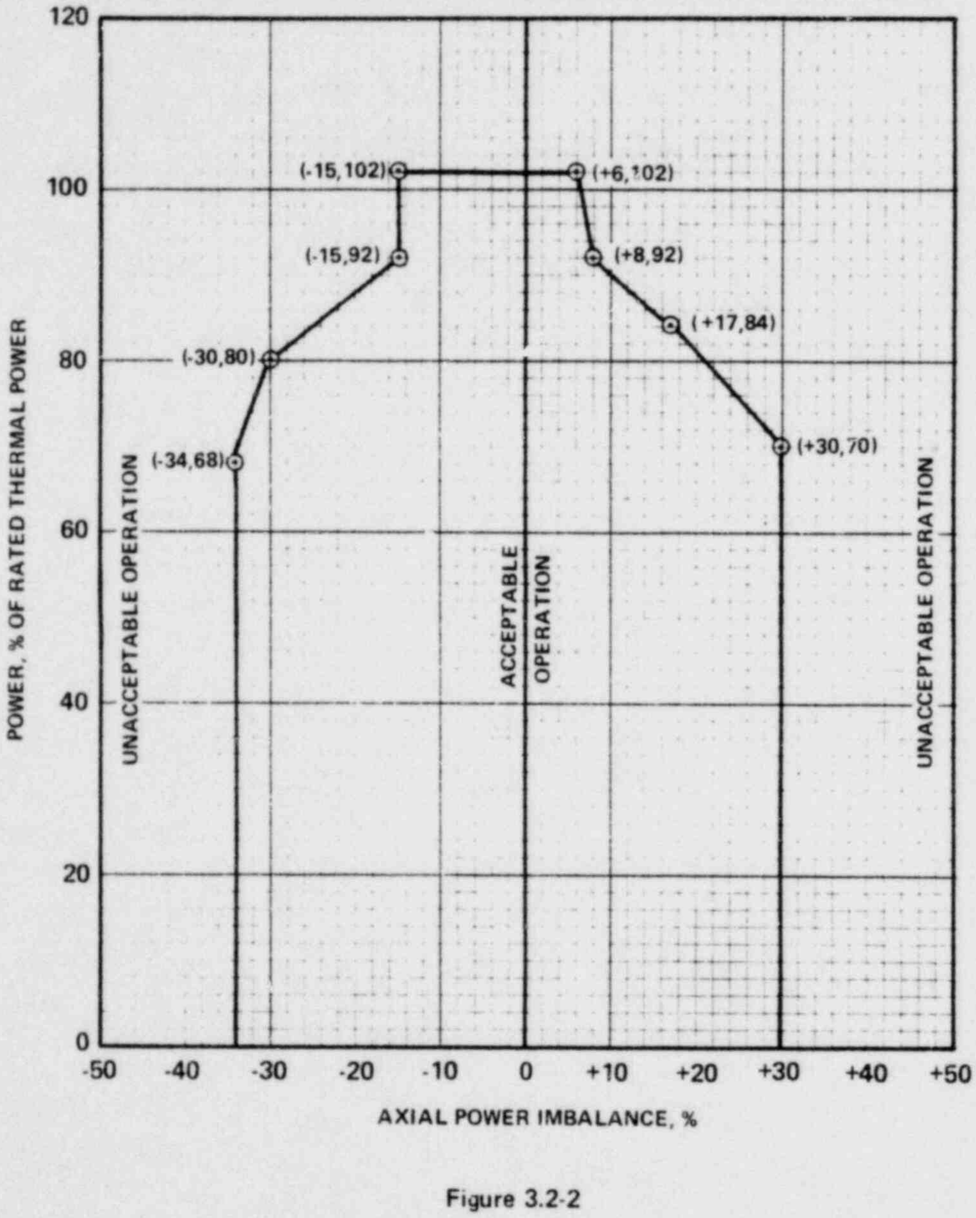


Figure 3.2-2

AXIAL POWER IMBALANCE Envelope for Operation After 400 ± 10 EFPD

POWER DISTRIBUTION LIMITS

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

$$F_Q \leq \frac{3.12}{P}$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$ and $P \leq 1.0$.

APPLICABILITY: MODE 1.

ACTION:

With F_Q exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% F_Q exceeds the limit within 15 minutes and similarly reduce the Nuclear Overpower Trip Setpoint and Nuclear Overpower based on RCS Flow and AXIAL POWER IMBALANCE Trip Setpoint within 4 hours.

b. Demonstrate through in-core mapping that F_Q is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that F_Q is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.2.1 F_Q shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

SURVEILLANCE REQUIREMENTS (Continued)

a. Prior to initial operation above 75 percent of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 The measured F_Q of 4.2.2.1 above, shall be increased by 1.4% to account for manufacturing tolerances and further increased by 7.5% to account for measurement uncertainty.

POWER DISTRIBUTION LIMITS

NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR - $F^N_{\Delta H}$

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \leq 1.78\ [1 + 0.6(1-P)]$$

where $P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and $P \leq 1.0$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% that $F^N_{\Delta H}$ exceeds the limit within 15 minutes and similarly reduce the Nuclear Overpower Trip Setpoint and Nuclear Overpower based on RCS Flow and AXIAL POWER IMBALANCE Trip Setpoint within 4 hours.

b. Demonstrate through in-core mapping that $F^N_{\Delta H}$ is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER above the reduced limit required by a or b, above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated through in-core mapping to be within its limit at a nominal 50% of RATED THERMAL POWER prior to exceeding this THERMAL POWER, at a nominal 75% of RATED THERMAL POWER prior to exceeding this THERMAL POWER and within 24 hours after attaining 95% or greater RATED THERMAL POWER.

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

4.2.3.1 F^N_{Δ} shall be determined to be within its limit by using the incore detectors to obtain a power distribution map:

a. Prior to operation above 75 percent of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

c. The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 5% for measurement uncertainty.

QUADRANT POWER TILT

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT shall not exceed the Steady State Limit of Table 3.2-2.

APPLICABILITY: MODE 1 above 15% of RATED THERMAL POWER.*

ACTION:

a. With the QUADRANT POWER TILT determined to exceed the Steady State Limit but less than or equal to the Transient Limit of Table 3.2-2.

   1. Within 2 hours:

      a) Either reduce the QUADRANT POWER TILT to within its Steady State Limit, or

      b) Reduce THERMAL POWER so as not to exceed THERMAL POWER, including power level cutoff, allowable for the reactor coolant pump combination less at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit and within 4 hours, reduce the Nuclear Overpower Trip Setpoint and the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE Trip Setpoint at least 2% for each 1% of QUADRANT POWER TILT in excess of the Steady State Limit.

   2. Verify that the QUADRANT POWER TILT is within its Steady State Limit within 24 hours after exceeding the Steady State Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the Nuclear Overpower Trip Setpoint to $\leq$ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

   3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

*See Special Test Exception 3.10.1.

LIMITING CONDITION FOR OPERATION (Continued)

b. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit of Table 3.2-2, due to misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce THERMAL POWER at least 2% for each 1% of indicated QUADRANT POWER TILT in excess of the Steady State Limit within 30 minutes.

2. Verify that the QUADRANT POWER TILT is within its Transient Limit within 2 hours after exceeding the Transient Limit or reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within the next 2 hours and reduce the Nuclear Overpower Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT determined to exceed the Transient Limit but less than the Maximum Limit of Table 3.2-2, due to causes other than the misalignment of either a safety, regulating or axial power shaping rod:

1. Reduce THERMAL POWER to less than 60% of THERMAL POWER allowable for the reactor coolant pump combination within 2 hours and reduce the Nuclear Overpower Trip Setpoint to ≤ 65.5% of THERMAL POWER allowable for the reactor coolant pump combination within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 60% of THERMAL POWER allowable for the reactor coolant pump combination may proceed provided that the QUADRANT POWER TILT is verified within its Steady State Limit at least once per hour for 12 hours or until verified at 95% or greater RATED THERMAL POWER.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. With the QUADRANT POWER TILT determined to exceed the Maximum Limit of Table 3.2-2, reduce THERMAL POWER to ≤ 15% of RATED THERMAL POWER within 2 hours.

SURVEILLANCE REQUIREMENTS

4.2.4. The QUADRANT POWER TILT shall be determined to be within the limits at least once every 7 days during operation above 15% of RATED THERMAL POWER except when the QUADRANT POWER TILT monitor is inoperable, then the QUADRANT POWER TILT shall be calculated at least once per 12 hours.

TABLE 3.2-2

QUADRANT POWER TILT LIMITS

	STEADY STATE LIMIT	TRANSIENT LIMIT	MAXIMUM LIMIT
Measurement Independent QUADRANT POWER TILT	4.92	11.07	20.0
QUADRANT POWER TILT as Measured by:			
Symmetrical Incore Detector System	4.01	9.51	20.0
Power Range Channels	2.04	7.04	20.0
Minimum Incore Detector System	2.82	8.32	20.0

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. Reactor Coolant Hot Leg Temperature

b. Reactor Coolant Pressure

c. Reactor Coolant Flow Rate

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 less than 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 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

Parameter	LIMITS		
	Four Reactor Coolant Pumps Operating	Three Reactor Coolant Pumps Operating	One Reactor Coolant Pump Operating in Each Loop
Reactor Coolant Hot Leg Temperature, T_H °F	$\leq$ 605.2	$\leq$ 605.2[1]	$\leq$ 605.2
Reactor Coolant Pressure, psig.[2]	$\geq$ 2062.7	$\geq$ 2058.9[1]	$\geq$ 2092.5
Reactor Coolant Flow Rate, gpm	$\geq$ 369,600	$\geq$ 276,091	$\geq$ 181,843

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the Reactor Protection System 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 Protection System 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 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 PROTECTION 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 PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	1	1	1	1, 2 and *	8
2. Nuclear Overpower	4	2	3	1, 2	2#
3. RCS Outlet Temperature--High	4	2	3	1, 2	3#
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	4	2(a)	3	1, 2	2#
5. RCS Pressure--Low	4	2(a)	3	1, 2	3#
6. RCS Pressure--High	4	2	3	1, 2	3#
7. Variable Low RCS Pressure	4	2(a)	3	1, 2	3#
8. Reactor Containment Pressure--High	4	2	3	1, 2	3#
9. Intermediate Range, Neutron Flux and Rate	2	0	2	1, 2 and *	4
10. Source Range, Neutron Flux and Rate					
A. Startup	2	0	2	2## and *	5
B. Shutdown	2	0	1	3, 4 and 5	6
11. Control Rod Drive Trip Breakers	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
12. Reactor Trip Module	2 per trip system	1 per trip system	2 per trip system	1, 2 and *	7#
13. Shutdown Bypass RCS Pressure-High	4	2	3	2**, 3**, 4**, 5**	6#

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above 10^{-10} amps on both Intermediate Range channels.

(a) Trip may be manually bypassed when RCS pressure $\leq$ 1720 psig by actuating Shutdown Bypass provided that:

(1) The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER,

(2) The Shutdown Bypass RCS Pressure--High Trip Setpoint of $\leq$ 1720 psig is imposed, and

(3) The Shutdown Bypass is removed when RCS pressure > 1800 psig.

ACTION STATEMENTS

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

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

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

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

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to ≤ 75% of RATED RATED THERMAL and the Nuclear Overpower Trip Setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours or the QUADRANT POWER TILT is monitored at least once per 12 hours.

ACTION 3 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

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

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1.

Action 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL Power level:

ACTION STATEMENTS (Continued)

a. $\leq$ 5% of RATED THERMAL POWER restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 5% of RATED THERMAL POWER.

b. > 5% of RATED THERMAL POWER, POWER OPERATION may continue.

ACTION 5 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:

a. $\leq 10^{-10}$ amps on the Intermediate Range (IR) instrumentation, restore the inoperable channel to OPERABLE status prior to increasing THERMAL POWER above 10^{-10} amps on the IR instrumentation.

b. $> 10^{-10}$ amps on the IR instrumentation, operation may continue.

ACTION 6 - 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 within one hour and at least once per 12 hours thereafter.

ACTION 7 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

a. Within 1 hour:

1. Place the inoperable channel in the tripped condition, or

2. Remove power supplied to the control rod trip device associated with the inoperative channel.

b. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1. The inoperable channel above may not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

ACTION 8 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIMES
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower*	$\leq$ 0.3 seconds
3. RCS Outlet Temperature--High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE*	$\leq$ 1.4 seconds
5. RCS Pressure--Low	$\leq$ 0.5 seconds
6. RCS Pressure--High	$\leq$ 0.5 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Reactor Containment Pressure--High	Not Applicable

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

TABLE 4.3-1

REACTOR PROTECTION SYSTEM 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. Nuclear Overpower	S	D(2) and Q(7)	M	1, 2
3. RCS Outlet Temperature--High	S	R	M	1, 2
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	S(4)	M(3) and Q(7,8)	M	1, 2
5. RCS Pressure--Low	S	R	M	1, 2
6. RCS Pressure--High	S	R	M	1, 2
7. Variable Low RCS Pressure	S	R	M	1, 2
8. Reactor Containment Pressure--High	S	R	M	1, 2
9. Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	1, 2 and *
10. Source Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	2, 3, 4 and 5
11. Control Rod Drive Trip Breaker	N.A.	N.A.	M and S/U(1)	1, 2 and *
12. Reactor Trip Module	N.A.	N.A.	M	1, 2, and *
13. Shutdown Bypass RCS Pressure-High	S	R	M	2**, 3**, 4**, 5**

TABLE 4.3-1 (Continued)

NOTATION

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

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

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - When THERMAL POWER [TP] is above 30% of RATED THERMAL POWER [RTP], compare out-of-core measured AXIAL POWER IMBALANCE [API_O] to incore measured AXIAL POWER IMBALANCE [API_I]. Recalibrate if:

$$\frac{RTP}{TP}[API_O - API_I] \geq 3.5\%$$

(4) - AXIAL POWER IMBALANCE and loop flow indications only.

(5) - Verify at least one decade overlap if not verified in previous 7 days.

(6) - Each train tested every other month.

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

(8) - Flow rate measurement sensors may be excluded from CHANNEL CALIBRATION. However, each flow measurement sensor shall be calibrated at least once per 18 months.

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

TABLE 3.3-3

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. SAFETY INJECTION					
a. High Pressure Injection					
1. Manual Initiation	2	1	2	1, 2, 3, 4	13
2. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
3. RCS Pressure Low	3	2	2	1, 2, 3*	9#
4. RCS Pressure Low-Low	3	2	2	1, 2, 3**	9#
5. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
b. Low Pressure Injection					
1. Manual Initiation	2	1	2	1, 2, 3, 4	13
2. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. RCS Pressure Low-Low	3	2	2	1, 2, 3**	9#
4. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10
2. REACTOR BLDG. COOLING AND ISOLATION					
a. Manual Initiation	2	1	2	1, 2, 3, 4	13
b. Reactor Bldg. Pressure High	3	2	2	1, 2, 3	9#
c. Automatic Actuation Logic	2	1	2	1, 2, 3, 4	10

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
3. REACTOR BLDG. SPRAY					
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	3	2	2	1, 2, 3	12
b. Automatic Actuation Logic	2	1	2	1, 2, 3	10
4. OTHER SAFETY SYSTEMS					
a. Reactor Bldg. Purge Isolation on High Radioactivity					
Gaseous	1	1	1	1, 2, 3, 4	11#

TABLE 3.3-3 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
b. Steam Line Rupture Matrix					
1. Low SG Pressure	2 per steam generator	1 per steam generator	2 per steam generator	1, 2, 3***	10
2. Automatic Actuation Logic	1 per steam generator	1 per steam generator	1 per steam generator	1, 2, 3	10

TABLE 3.3-3 (Continued)

TABLE NOTATION

*Trip function may be bypassed in this MODE with RCS pressure below 1700 psig. Bypass shall be automatically removed when RCS pressure exceeds 1700 psig.

**Trip function may be bypassed in this MODE with RCS pressure below 900 psig. Bypass shall be automatically removed when RCS pressure exceeds 900 psig.

***Trip function may be bypassed in this MODE with steam generator pressure below 725 psig. Bypass shall be automatically removed when steam generator pressure exceeds 765 psig.

#The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 9 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed until performance of the next required CHANNEL FUNCTIONAL TEST provided the inoperable channel is placed in the tripped condition within 1 hour.

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours; however, one channel may be bypassed for up to 1 hour for surveillance testing per Specification 4.3.2.1.1.

ACTION 11 - With less than the Minimum Channels OPERABLE, operation may continue provided the containment purge and exhaust valves are maintained closed.

ACTION 12 - With the number of OPERABLE Channels one less than the Total Number of Channels operation may proceed provided the inoperable channel is placed in the bypassed condition and the minimum channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for Surveillance testing per Specification 4.3.2.1.

ACTION 13 - 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.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION		
a. High Pressure Injection ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	≤ 4 psig	≤ 4 psig
3. RCS Pressure Low	≥ 1500 psig	≥ 1500 psig
4. RCS Pressure Low-Low	≥ 500 psig	≥ 500 psig
5. Automatic Actuation Logic	Not Applicable	Not Applicable
b. Low Pressure Injection ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	≤ 4 psig	≤ 4 psig
3. RCS Pressure Low-Low	≥ 500 psig	≥ 500 psig
4. Automatic Actuation Logic	Not Applicable	Not Applicable
2. REACTOR BLDG. COOLING & ISOLATION		
a. ES Actuation "A" and "B"		
1. Manual Initiation	Not Applicable	Not Applicable
2. Reactor Bldg. Pressure High	≤ 4 psig	≤ 4 psig
3. Automatic Actuation Logic	Not Applicable	Not Applicable
b. ES Actuation Indication "AB"		
1. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-4 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
3. REACTOR BLDG. SPRAY		
a. Reactor Bldg. Pressure High-High	$\leq$ 30 psig	$\leq$ 30 psig
coincident with HPI Signal	See 1.a.2, 3, 4	See 1.a.2, 3, 4
b. Automatic Actuation Logic	Not Applicable	Not Applicable
4. OTHER SAFETY SYSTEMS		
a. Reactor Bldg. Purge Isolation on High Radioactivity		
Gaseous	1×10^2 μci/sec	Not Applicable
b. Steam Line Rupture Matrix		
1. Low SG Pressure	$\geq$ 600 psig	$\geq$ 600 psig
2. Automatic Actuation Logic	Not Applicable	Not Applicable

TABLE 3.3-5

ENGINEERED SAFETY FEATURES RESPONSE TIMES

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS*
1. Manual	
a. High Pressure Injection	Not Applicable
b. Low Pressure Injection	Not Applicable
c. Reactor Building Cooling	Not Applicable
d. Reactor Building Isolation	Not Applicable
e. Reactor Building Spray	Not Applicable
f. Reactor Building Purge Isolation	Not Applicable
g. Steam Line Rupture Matrix	
1) Emergency Feedwater Actuation	Not Applicable
2) Feedwater Isolation	Not Applicable
3) Steam Line Isolation	Not Applicable
2. Reactor Building Pressure-High	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*
c. Reactor Building Cooling	25*
d. Reactor Building Isolation	6[illegible]*
3. Reactor Building Pressure High-High (with HPI signal)	
a. Reactor Building Spray	56*
4. RCS Pressure Low	
a. High Pressure Injection	25*
5. RCS Pressure Low-Low	
a. High Pressure Injection	25*
b. Low Pressure Injection	25*
6. Low Steam Generator Pressure	
a. Feedwater Isolation	34
b. Steam Line Isolation	5

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

TABLE 4.3-2

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1. SAFETY INJECTION				
a. High Pressure Injection				
1. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
2. Reactor Bldg. Pressure High	S	R	M(2)	1, 2, 3
3. RCS Pressure Low	S	R	M	1, 2, 3
4. RCS Pressure Low-Low	S	R	M	1, 2, 3
5. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4
b. Low Pressure Injection				
1. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
2. Reactor Bldg. Pressure High	S	R	M(2)	1, 2, 3

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. RCS Pressure Low-Low	S	R	M	1, 2, 3
4. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4
2. REACTOR BLDG. COOLING AND ISOLATION				
a. Manual Initiation	N/A	N/A	M(1)	1, 2, 3, 4
b. Reactor Bldg. Pressure High	S	R	M(2)	1, 2, 3
c. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3, 4

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BLDG. SPRAY				
a. Reactor Bldg. Pressure High-High coincident with HPI Signal	S	R	M(4)	1, 2, 3
b. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3
4. OTHER SAFETY SYSTEMS				
a. Reactor Bldg. Purge Isolation on High Radioactivity				
1. Gaseous	S	R	M	All Modes
b. Steam Line Rupture Matrix				
1. Low SG Pressure	N/A	R	N/A	1, 2, 3
2. Automatic Actuation Logic	N/A	N/A	M(3)	1, 2, 3

TABLE 4.3-2 (Continued)

TABLE NOTATION

(1) Manual actuation switches shall be tested at least once per 18 months during shutdown. All other circuitry associated with manual safeguards actuation shall receive a CHANNEL FUNCTIONAL TEST at least once per 31 days.

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

(3) Each logic channel shall be tested at least every other 31 days.

(4) Reactor Bldg. Pressure High-High signal only.

3/4.3.3 MONITORING INSTRUMENTATION

RADIATION MONITORING INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: As shown in Table 3.3-6.

ACTION:

a. With a radiation monitoring channel alarm/trip setpoint exceeding the value shown in Table 3.3-6, adjust the setpoint to within the limit within 2 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/TRIP SETPOINT	MEASUREMENT RANGE	ACTION
1. AREA MONITORS					
a. Fuel Storage Pool Area					
i. Criticality Monitor	1	*	≤ 15 mR/hr	10^{-1} - 10^{4} mR/hr	14
2. PROCESS MONITORS					
a. Fuel Storage Pool Area					
i. Gaseous Activity - Ventilation System Isolation	1	**	≤ 2 x background	10^{1} - 10^{6} cpm	16
b. Containment					
i. Gaseous Activity -					
a) Purge & Exhaust Isolation	1	6	≤ 2 x background	10^{1} - 10^{6} cpm	17
b) RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{1} - 10^{6} cpm	15
ii. Iodine Activity - RCS Leakage Detection	1	1, 2, 3 & 4	Not Applicable	10^{1} - 10^{6} cpm	15

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool

TABLE 3.3-6 (Continued)

TABLE NOTATION

ACTION 14 - 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 15 - 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 3.9.12.

ACTION 17 - 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				
i. Criticality Monitor	S	R	M	*
2. PROCESS MONITORS				
a. Fuel Storage Pool Area				
i. Gaseous Activity - Ventilation System Isolation	S	R	M	**
b. Containment				
i. Gaseous Activity -				
a) Purge & Exhaust Isolation	S	R	M	6
b) RCS Leakage Detection	S	R	M	1, 2, 3, & 4
ii. Iodine Activity - RCS Leakage Detection	S	R	M	1, 2, 3, & 4

* With fuel in the storage pool or building
** With irradiated fuel in the storage pool

INSTRUMENTATION

INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

3.3.3.2 As a minimum, the incore detectors shall be OPERABLE as specified below.

a. For AXIAL POWER IMBALANCE measurements:

1. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane with one plane at the core mid-plane and one plane in each axial core half.

2. The axial planes in each core half shall be symmetrical about the core mid-plane.

3. The detector strings shall not have radial symmetry.

b. For QUADRANT POWER TILT measurements with the Minimum Incore Detector System:

1. Two sets of 4 detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

2. Detectors in the same plane shall have quarter core radial symmetry.

c. For QUADRANT POWER TILT measurements with the Symmetric Incore Detector System at least 75% of the detectors in each core quadrant shall be OPERABLE.

APPLICABILITY: When the incore detection system is used for surveillance of:

a. The AXIAL POWER IMBALANCE, or

b. The QUADRANT POWER TILT.

ACTION:

With less than the specified minimum incore detector arrangement OPERABLE, do not use incore detector measurements to determine AXIAL POWER IMBALANCE or QUADRANT POWER TILT. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.3.3.2 The incore detector system shall be demonstrated OPERABLE:

a. By performance of a CHANNEL CHECK within 7 days prior to its use for measurement of the AXIAL POWER IMBALANCE or the QUADRANT POWER TILT.

b. At least once per 18 months by performance of a CHANNEL CALIBRATION which does not include the neutron detectors.

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

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

SURVEILLANCE REQUIREMENTS

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

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

TABLE 3.3-7

SEISMIC MONITORING INSTRUMENTATION

INSTRUMENTS AND SENSOR LOCATIONS	MEASUREMENT RANGE	MINIMUM INSTRUMENT OPERABLE
1. Triaxial Time-History Accelographs		
a. 95'0" Containment vessel foundation	± 1.0 G	1
b. 267'6" Outside containment on top of ring girder	± 1.0 G	1
c. 145'0" Control room floor	± 1.0 G	1
2. Triaxial Peak Accelographs		
a. 140'0" At top of reactor	± 2.0 G	1
b. 175'6" Piping at top of one S.G.	± 2.0 G	1
c. 156'8" Top of Borated Water Storage Tank	± 2.0 G	1
3. Triaxial Seismic Switches		
a. 95'0" Containment vessel foundation	.005 to .05 G	1*

NOTE: Starts all three magnetic time-history accelographs whenever the acceleration exceeds .01 ^

* With reactor control room indication

TABLE 4.3-4

SEISMIC MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTS AND SENSOR LOCATIONS	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST
1. Triaxial Time-History Accelographs			
a. 95'0" Containment vessel foundation	M*	R	SA
b. 267'6" Outside Containment on top of ring girder	M*	R	SA
c. 145'0" Control room floor	M*	R	SA
2. Triaxial Peak Accelographs			
a. 140'0" At top of reactor	R	NA	NA
b. 175'6" Piping at top of one S.G.	R	NA	NA
c. 166'8" Top of Borated Water Storage Tank	R	NA	NA
3. Triaxial Seismic Switches			
a. 95'0" Containment vessel foundation	M**	R	SA

* Except seismic trigger
** With reactor control room indication

INSTRUMENTATION

METEOROLOGICAL INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: At all times.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-8

METEOROLOGICAL MONITORING INSTRUMENTATION

INSTRUMENT	LOCATION	MINIMUM OPERABLE
1. WIND SPEED		
a. Nominal Elev. 175'		1
b. Nominal Elev. 33'		1
2. WIND DIRECTION		
a. Nominal Elev. 175'		1
b. Nominal Elev. 33'		1
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 175'		1
b. Nominal Elev. 33'		1

TABLE 4.3-5

METEOROLOGICAL MONITORING INSTRUMETATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. WIND SPEED		
a. Nominal Elev. 175'	D	SA
b. Nominal Elev. 33'	D	SA
2. WIND DIRECTION		
a. Nominal Elev. 175'	D	SA
b. Nominal Elev. 33'	D	SA
3. AIR TEMPERATURE - DELTA T		
a. Nominal Elev. 175'	D	SA
b. Nominal Elev. 33'	D	SA

INSTRUMENTATION

REMOTE SHUTDOWN INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

INSTRUMENT		READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1.	Reactor Trip Breaker Indication	CRD switch gear room 124 foot elevation	open-close	1 per trip breaker and 1 per secondary trip breaker
2.	Reactor Coolant Temperature - Th	4160ES-B switchgear room 108 foot elevation	520-620°F	1 per loop
3.	Reactor Coolant Pressure	4160ES-B switchgear room 108 foot elevation	0-2500 psig	1
4.	Pressurizer Level	4160ES-B switchgear room 108 Foot elevation	0-320" H_2O	1
5.	Steam Generator Pressure	4160ES-B switchgear room 108 foot elevation	0-1200 psig	1 per steam generator
6.	Steam Generator Level	4160ES-B switchgear room 108 foot elevation	0-250" H_2O	1 per steam generator
7.	Decay Heat Closed Cycle Cooling Temperature	4160ES-B switchgear room 108 foot elevation	0-300°F	1 per cooler
8.	Motor driven Emergency Feedwater Pressure	Intermediate Building 95 foot elevation	0-2000 psig	1 per pump
9.	Nuclear Services Closed Cycle Cooling Pumps Discharge Pressure	Auxiliary Building 95 foot elevation	0-300 psig	1
10.	Nuclear Services Closed Cycle Cooling Cooler Outlet Temperature	Auxiliary Building 95 foot elevation	0-250°F	1 per cooler

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Trip Breaker Indication	M	N.A.
2. Reactor Coolant Temperature - T_h	M	R
3. Reactor Coolant Pressure	M	R
4. Pressurizer Level	M	R
5. Steam Generator Level	M	R
6. Steam Generator Pressure	M	R
7. Decay Heat Closed Cycle Cooling Temperature	M	R
8. Motor Driven Emergency Feedwater Pressure	M	R
9. Nuclear Services Closed Cycle Cooling Pumps Discharge Pressure	M	R
10. Nuclear Services Closed Cycle Cooling Cooler Outlet Temperature	M	R

INSTRUMENTATION

POST-ACCIDENT INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3.6 The post-accident monitoring instrumentation channels shown in Table 3.3-10 shall be OPERABLE with readouts and recorders in the control room.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With the number of OPERABLE post-accident monitoring channels less than required by Table 3.3-10, either restore the inoperable channel to OPERABLE status within 30 days, or be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.3-10

POST-ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Power Range Nuclear Flux	0-125%	2
2. Reactor Building Pressure	0-70 psia	2
3. Source Range Nuclear Flux	10^{-1} to 10^{6} cps	2
4. Reactor Coolant Outlet Temperature	520-620°F	2 per loop
5. Reactor Coolant Total Flow	0-110% full flow	1
6. RC Loop Pressure	0-2500, 0-500, and 1700-2500 psig	2 each range
7. Pressurizer Level	0-320 inches	2
8. Steam Generator Outlet Pressure	0-1200 psig	2/steam generator
9. Steam Generator Operating Range Level	0-100%	2/steam generator
10. Borated Water Storage Tank Level	0-50 feet	2
11. Startup Feedwater Flow	0-1.5x10^{6} lb/hr.	2

TABLE 4.3-7

POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Power Range Nuclear Flux	M	Q*
2. Reactor Building Pressure	M	R
3. Source Range Nuclear Flux	M	R*
4. Reactor Coolant Outlet Temperature	M	R
5. Reactor Coolant Total Flow Rate	M	R
6. RC Loop Pressure	M	R
7. Pressurizer Level	M	R
8. Steam Generator Outlet Pressure	M	R
9. Steam Generator Level	M	R
10. Borated Water Storage Tank Level	M	R
11. Startup Feedwater Flow Rate	M	R

*Neutron detectors may be excluded from CHANNEL CALIBRATION.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Both reactor coolant loops and both reactor coolant pumps in each loop shall be in operation.

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

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 78% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

   1. Nuclear Overpower

b. With one reactor coolant pump in each loop not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 51.2% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with one reactor coolant pump operating in each loop:

   1. Nuclear Overpower

*See Special Test Exception 3.10.3.

LIMITING CONDITION FOR OPERATION (Continued)

MODES 3 ,4 and 5:

a. Operation may proceed provided at least one reactor coolant loop is in operation with an associated reactor coolant pump or decay heat removal pump.

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

SURVEILLANCE REQUIREMENTS

4.4.1 The Reactor Protective Instrumentation channels specified in the applicable ACTION statement above shall be verified to have had their trip setpoints changed to the values specified in Specification 2.2.1 for the applicable number of reactor coolant pumps operating either:

a. Within 4 hours after switching to a different pump combination if the switch is made while operating, or

b. Prior to reactor criticality if the switch is made while shutdown.

REACTOR COOLANT SYSTEM

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 PSIG $\pm$ 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 DHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

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

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 PSIG ± 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 No additional Surveillance Requirements other than those required by Specification 4.0.5.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.4 The pressurizer shall be OPERABLE with:

a. A steam bubble,

b. A water level between 40 and 290 inches.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.4 The pressurizer shall be demonstrated OPERABLE by verifying pressurizer level to be within limits at least once per 12 hours.

STEAM GENERATORS

LIMITING CONDITION FOR OPERATION

3.4.5 Each steam generator shall be OPERABLE with a water level between 18 and 360 inches.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

b. With one or more steam generators inoperable due to the water level being outside the limits, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

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:

SURVEILLANCE REQUIREMENTS (Continued)

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

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

4.4.5.6 The steam generator shall be demonstrated OPERABLE by verifying steam generator level to be within limits at least once per 12 hours.

TABLE 4.4–1

MINIMUM NUMBER OF STEAM GENERATORS TO BE
INSPECTED DURING INSERVICE INSPECTION

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

Table Notation:

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

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

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

TABLE 4.4–2

STEAM GENERATOR TUBE INSPECTION

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

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

REACTOR COOLANT SYSTEM

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. The containment atmosphere iodine radioactivity monitoring system,

b. The containment sump level monitoring system, and

c. The containment atmosphere gaseous radioactivity monitoring system.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.6.1 The leakage detection systems shall be demonstrated OPERABLE by:

SURVEILLANCE REQUIREMENTS (Continued)

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

b. Containment sump level monitoring system-performance of CHANNEL CALIBRATION at least once per 18 months.

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

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.6.2 Reactor Coolant System leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE,

b. 1 GPM UNIDENTIFIED LEAKAGE,

c. 1 GPM total primary-to-secondary leakage through steam generators, and

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

e. 10 GPM CONTROLLED LEAKAGE at a Reactor Coolant System pressure of 2150 $\pm$ 20 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

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 iodine radioactivity monitor at least once per 12 hours,

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

SURVEILLANCE REQUIREMENTS (Continued)

c. Measurement of the CONTROLLED LEAKAGE from the reactor coolant pump seals with the modulating valve fully open when the Reactor Coolant System pressure is 2150 $\pm$ 20 psig at least once per 31 days,

d. Performance of a Reactor Coolant System water inventory balance at least once per 72 hours during steady state operation.

CHEMISTRY

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: At all times.

ACTION:

MODES 1, 2, 3 and 4.

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

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

At all other times

With the concentration of either chloride or fluoride in the Reactor Coolant System in excess of its Steady State Limit for more than 24 hours or in excess of its Transient Limit, reduce the Reactor Coolant System pressure to $\leq$ 500 psig, if applicable, and perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation prior to increasing the system pressure above 500 psig 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	$\leq$ 0.15 ppm	$\leq$ 1.50 ppm

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

TABLE 4.4-3

REACTOR COOLANT SYSTEM

CHEMISTRY LIMITS SURVEILLANCE REQUIREMENTS

PARAMETER	SAMPLE AND ANALYSIS FREQUENCY
DISSOLVED OXYGEN*	At least once each 72 hours
CHLORIDE	At least once each 72 hours
FLUORIDE	At least once each 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 48 hours provided that operation under these circumstances shall not exceed 10% 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 48 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in at least HOT STANDBY with T_{avg} < 500°F within 6 hours.

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

MODES 1, 2, 3, 4 and 5:

a. 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:

*With $T_{avg} \geq$ 500°F.

ACTION: (Continued)

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

2. Fuel burnup by core region,

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

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

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

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

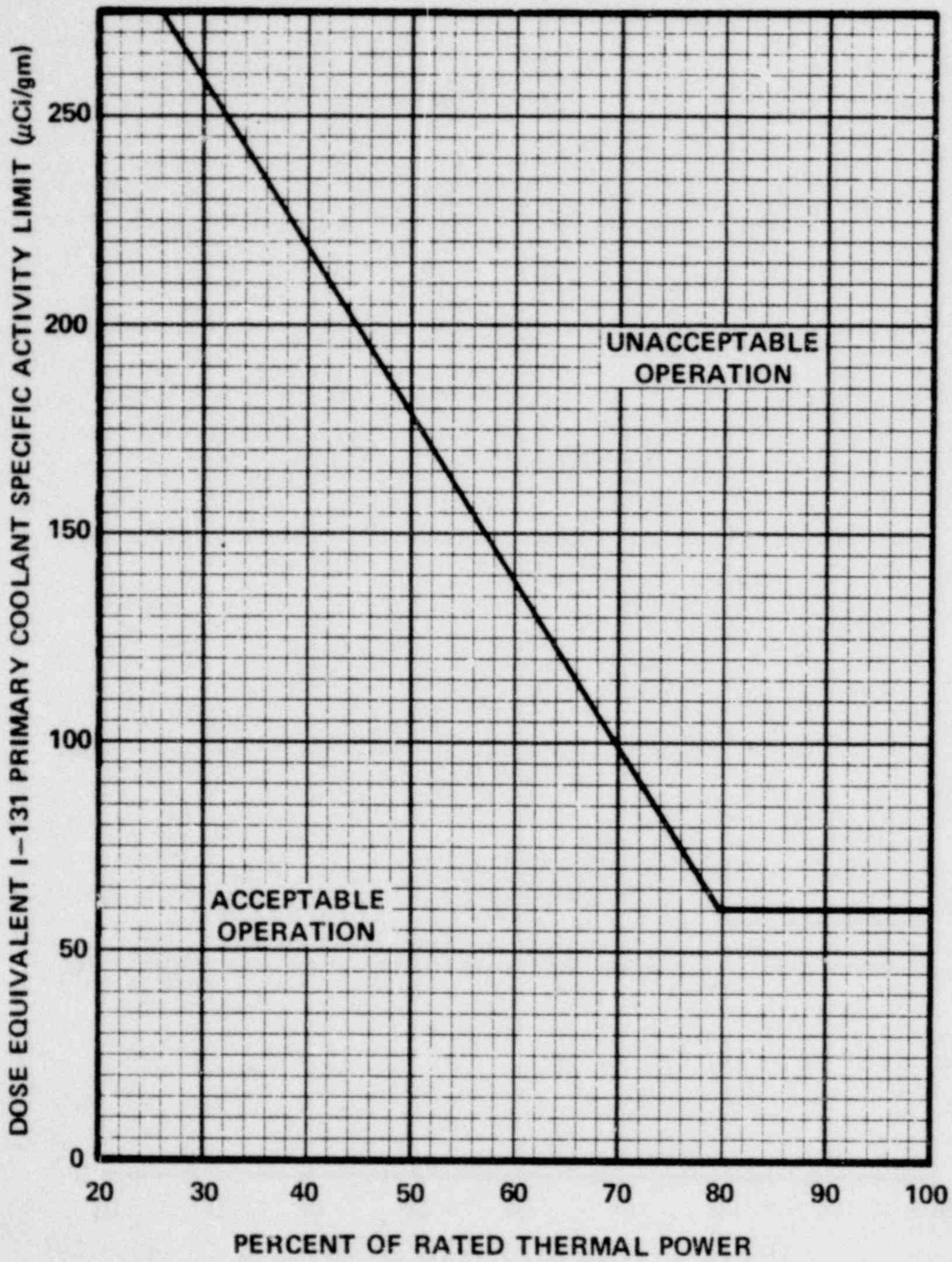


FIGURE 3.4-1

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

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2, 3.4-3, and 3.4-4 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing 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 $\leq$ 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 engineering evaluation 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 operation or be in at least HOT STANDBY within the next 6 hours and reduce RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

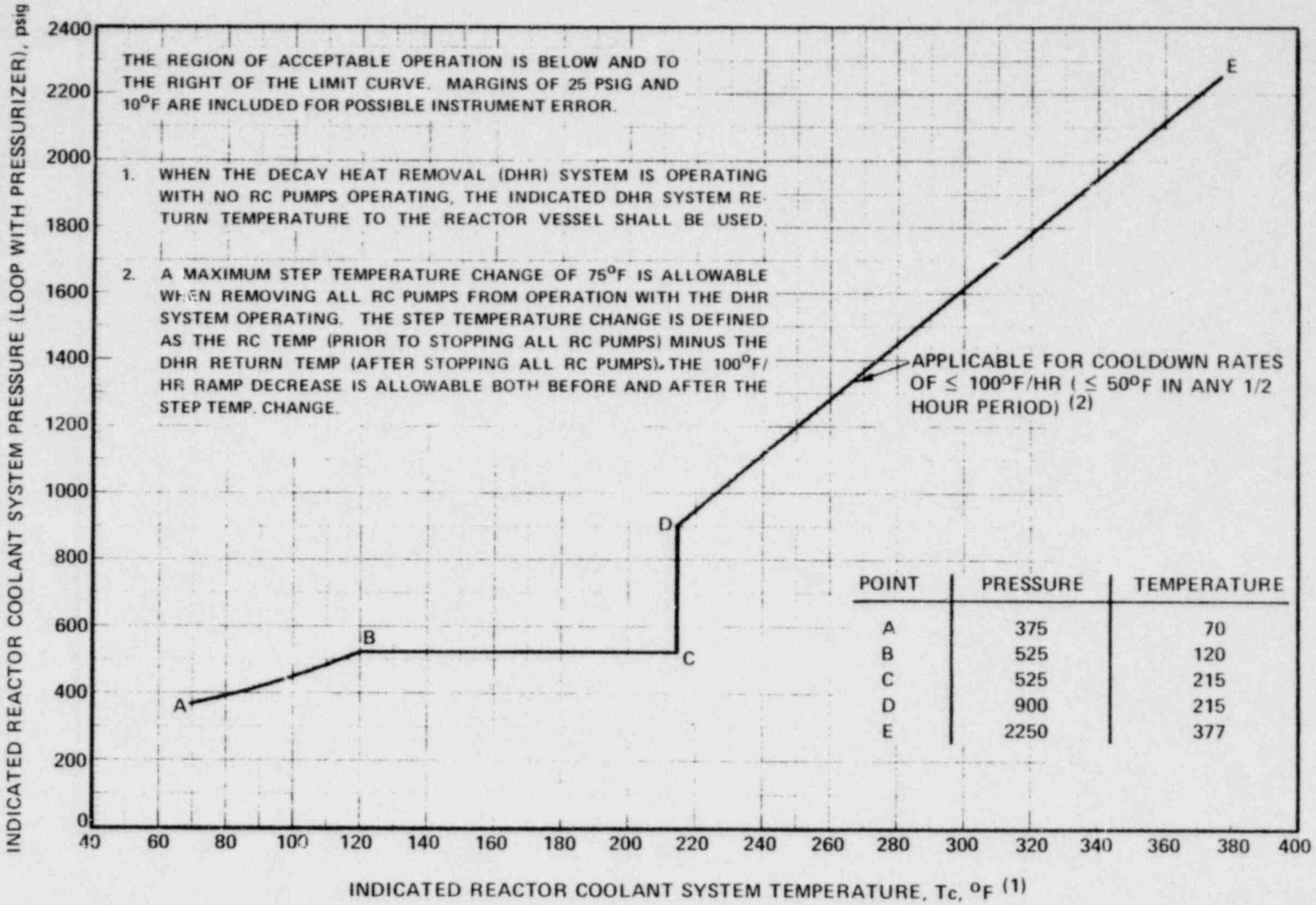


Figure 3.4-3 Reactor Coolant System Pressure—Temperature Limits
For Cooldown for the First 5 EFPY

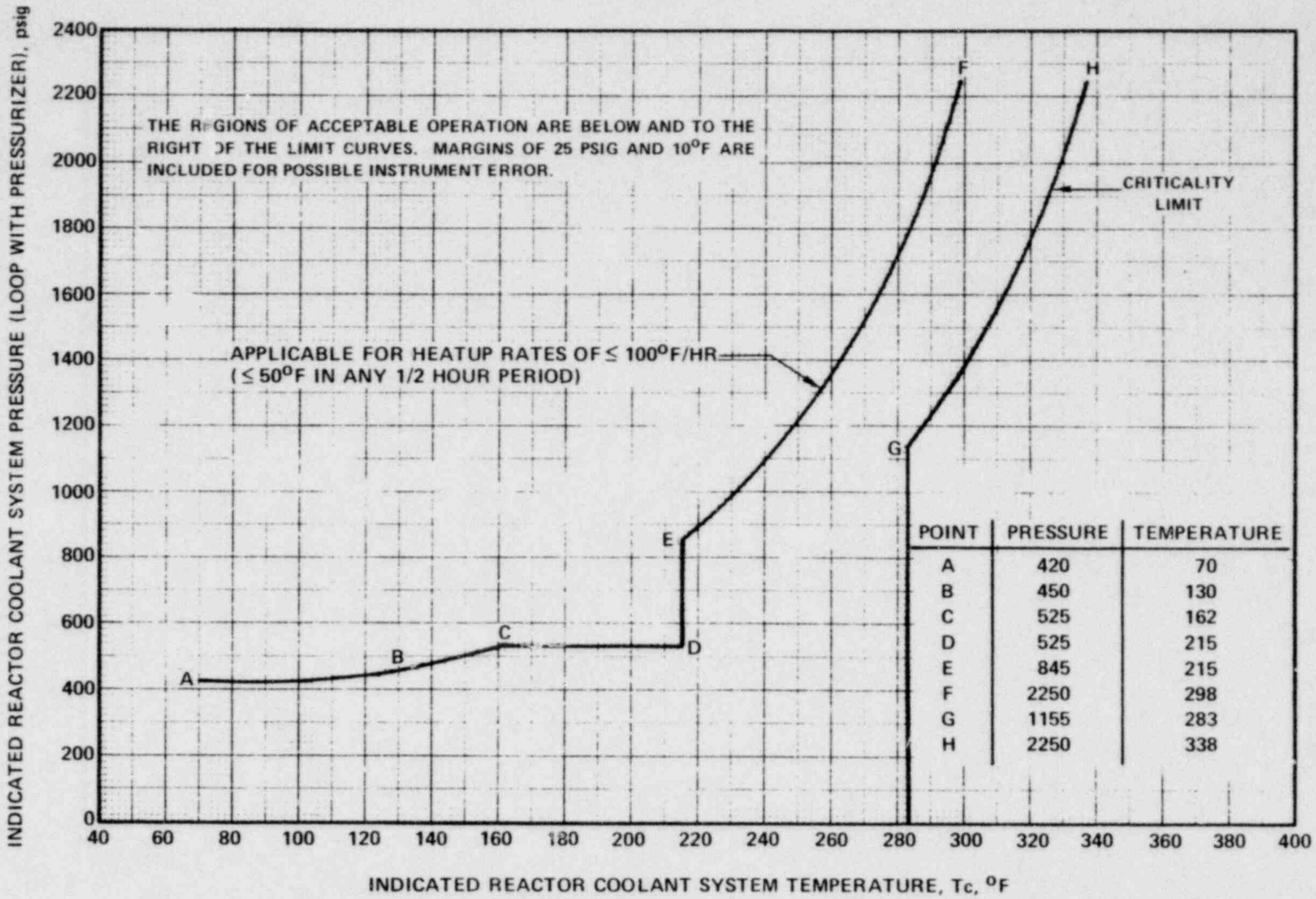
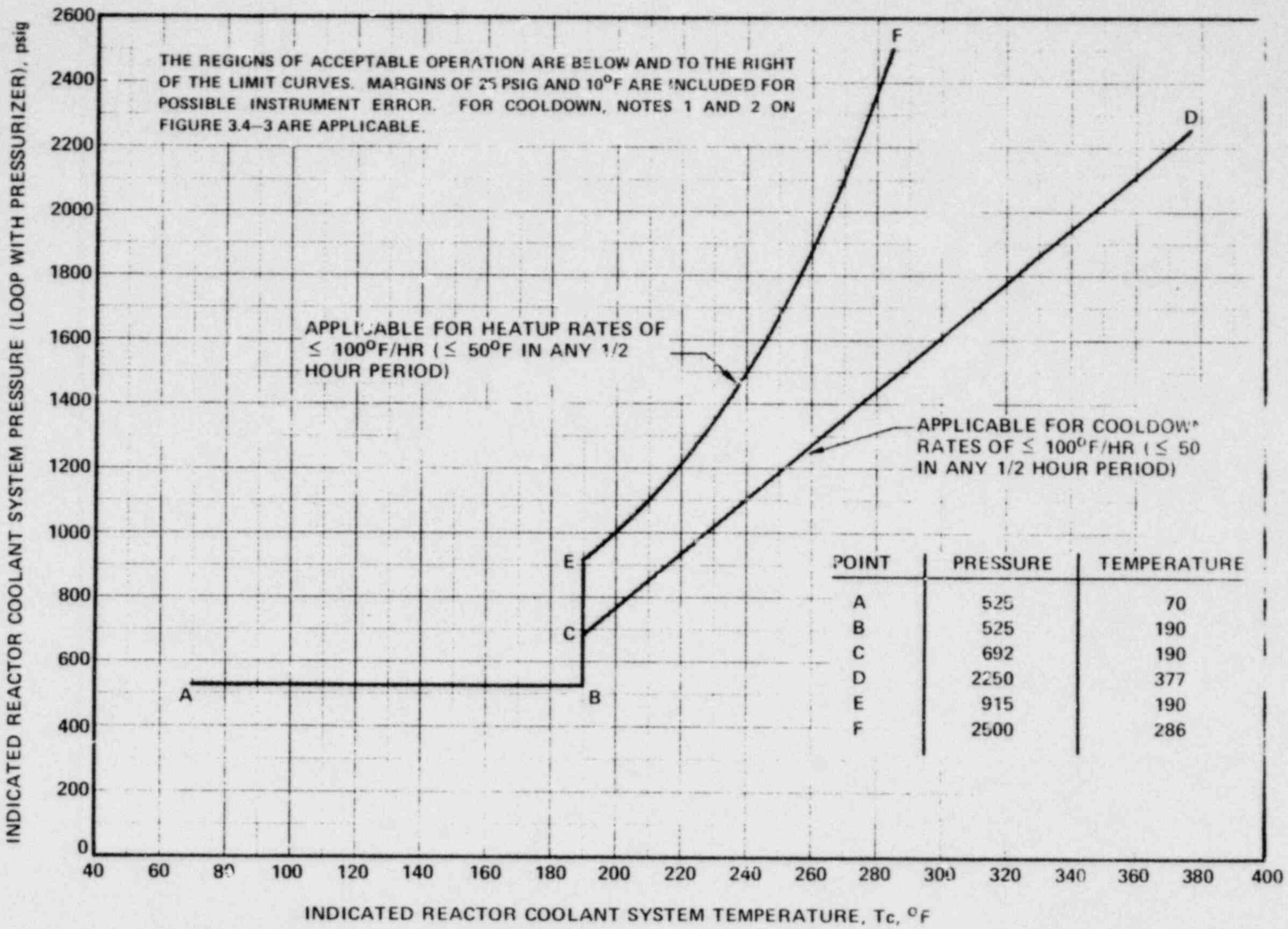


Figure 3.4-2 Reactor Coolant System Pressure–Temperature Limits for Heatup and Core Criticality for the First 5 EFPY

SURVEILLANCE REQUIREMENTS

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

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, at the intervals shown in Table 4.4-5. The results of these examinations shall be used to update Figures 3.4-2, 3.4-3 and 3.4-4.



POINT	PRESSURE	TEMPERATURE
A	525	70
B	525	190
C	692	190
D	2250	377
E	915	190
F	2500	286

Figure 3.4-4 Reactor Coolant System Pressure - Temperature Heatup and Cooldown Limits for Inservice Leak and Hydrostatic Tests for the First 5 EFPY

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Capsule	Installation	Removal
A	End of First Cycle	Standby
C	End of First Cycle	End of Ninth Cycle
E	End of First Cycle	Standby
B	Initial Fuel Load	End of First Cycle
D	Initial Fuel Load	End of Fifth Cycle
F	End of First Cycle	End of Fifth Cycle

REACTOR COOLANT SYSTEM

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

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

b. A maximum spray water temperature differential of 410°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within limits within 30 minutes; perform an engineering evaluation 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 pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperature 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 once per 12 hours during auxiliary spray operation with pressurizer temperature > 440°F.

REACTOR COOLANT SYSTEM

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: All MODES.

ACTION:

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

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

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1 In addition to the requirements of Specification 4.0.5:

a. The reactor coolant pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b. of Regulatory Guide 1.14, Revision 1, August 1975.

SURVEILLANCE REQUIREMENTS (Continued)

b. Each internals vent valve shall be demonstrated OPERABLE at least once per 18 months during shutdown, by:

1. Verifying through visual inspection that the valve body and valve disc exhibit no abnormal degradation,

2. Verifying the valve is not stuck in an open position, and

3. Verifying through manual actuation that the valve begins to open with force a equivalent to $\leq$ 0.15 psid and is fully open with a force equivalent to $\leq$ 0.30 psid.

CORE FLOODING TANKS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system core flooding tank shall be OPERABLE with:

a. The isolation valve open,

b. A contained borated water volume between 7555 and 8005 gallons of borated water,

c. Between 2270 and 3500 ppm of boron, and

d. A nitrogen cover-pressure of between 575 and 625 psig.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

a. With one core flooding 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 12 hours.

b. With any core flooding 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 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each core flooding tank shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each tank isolation valve is open.

*With Reactor Coolant pressure >750 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours of each solution volume increase of $\geq$ 80 gallons by verifying the boron concentration of the tank solution.

c. At least once per 31 days by verifying that power to the isolation valve operator is removed by locking the breaker in the open position.

d. At least once per 18 months by verifying that each core flooding tank isolation valve closed alarm actuates whenever each core flooding tank isolation valve is not fully open and the Reactor Coolant System pressure exceeds 750 psig.

EMERGENCY CORE COOLING SYSTEMS

ECCS SUBSYSTEMS - T_{avg} > 280°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 injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) on a safety injection signal and manually transferring suction to the containment sump during the recirculation phase of operation.

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.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

b. 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 suction 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 each containment entry when CONTAINMENT INTEGRITY is established.

c. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the DHR system from the Reactor Coolant System when the Reactor Coolant System pressure is $\geq$ 284 psig.

SURVEILLANCE REQUIREMENTS (Continued)

2. A visual inspection of the containment emergency sump which verifies 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 a total leak rate $\leq$ 6 gallons per hour for the LPI system at:

a) Normal operating pressure or hydrostatic test pressure $\geq$ 150 psig for those parts of the system downstream of the pump suction isolation valve, and

b) $\geq$ 55 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on a high pressure or low pressure safety injection test signal, as appropriate.

2. Verifying that each HPI and LPI pump test starts automatically upon receipt of a high pressure or low pressure safety injection test signal, as appropriate.

ECCS SUBSYSTEMS - T_{avg} < 280°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 injection (HPI) pump,

b. One OPERABLE low pressure injection (LPI) pump,

c. One OPERABLE decay heat cooler, and

d. An OPERABLE flow path capable of taking suction from the borated water storage tank (BWST) and transferring suction to the containment emergency sump.

APPLICABILITY: MODE 4.

ACTION:

a. With no ECCS subsystem OPERABLE because of the inoperability of either the HPI pump or the flow path from the borated water storage tank, restore at least one ECCS subsystem to OPERABLE status within one hour or be in COLD SHUTDOWN within the next 20 hours.

b. With no ECCS subsystem OPERABLE because of the inoperability of either the decay heat cooler or LPI pump, restore at least one ECCS subsystem to OPERABLE status or maintain the Reactor Coolant System T_{avg} less than 280°F by use of alternate heat removal methods.

c. 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 subsystems shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.5.2.

BORATED WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 The borated water storage tank (BWST) shall be OPERABLE with:

a. A contained borated water volume of between 415,200 and 449,000 gallons,

b. Between 2270 and 2450 ppm of boron, and

c. A minimum water temperature of 40°F

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.5.4 The BWST shall be demonstrated OPERABLE:

a. At least once per 7 days by:

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

2. Verifying the boron c ncentration of the water.

b. At least once per 24 hours by verifying the water temperature when outside air temperature <40°F.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

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

   1. All penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except as provided in Table 3.6-1 of Specification 3.6.3.1, and

   2. All equipment hatches are closed and sealed.

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

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

CONTAINMENT LEAKAGE

LIMITING CONDITION FOR OPERATION

3.6.1.2 Containment leakage rates shall be limited to:

a. An overall integrated leakage rate of $\leq L_a$, 0.25 percent by weight of the containment air per 24 hours at P_a, 49.6 psig.

b. A combined leakage rate of $\leq 0.60\ L_a$ for all penetrations and valves subject to Type B and C tests, when pressurized to P_a.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With either (a) the measured overall integrated containment leakage rate exceeding $0.75\ L_a$ or (b) with the measured combined leakage rate for all penetrations and valves subject to Type B and C tests exceeding $0.60\ L_a$, 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 provisions 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 P_a, 49.6 psig, during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

SURVEILLANCE REQUIREMENTS (Continued)

b. If any periodic Type A test fails to meet .75 L_a 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 .75 L_a a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet .75 L_a 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.

   2. Had 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, 49.6 psig.

d. Type B and C tests shall be conducted with gas at P_a, 49.6 psig at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Penetrations using continuous leakage monitoring systems, and

   3. Valves pressurized with fluid from a seal system.

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

SURVEILLANCE REQUIREMENTS (Continued)

f. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J, Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a, 54.6 psig, and the seal system capacity is adequate to maintain system pressure for at least 30 days.

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

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, 49.6 psig.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With an air lock inoperable, 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.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each containment air lock shall be demonstrated OPERABLE:

a.* After each opening, except when the air lock is being used for multiple entries, then at least once per 72 hours, by verifying seal leakage $\leq 0.01\ L_a$ when the volume between the door seals is pressurized to ≥ 8 psig, for at least 30 seconds.

b. At least once per 6 months by conducting an overall air lock leakage test at P_a, 49.6 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.

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

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between 17.7 and 12.7 psia.

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 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 130°F.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the containment average air temperature > 130°F, reduce the average air temperature to within the limit within 8 hours, or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

Location

a. Column RB-320, elevation 100'

b. Colume RB-320, elevation 125'

c. Outside secondary shield wall, elevation 180'

d. Crane access platform, elevation 235'

CONTAINMENT STRUCTURAL INTEGRITY

LIMITING CONDITIONS FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

a. Determining that a representative sample of at least 21 tendons (6 dome, 5 vertical, and 10 hoop) each have a lift off force of between 1,249,000 (minimum) and 1,721,000 (maximum) pounds. This test shall include an unloading cycle in which each of these tendons is detensioned to determine if any wires or strands are broken or damaged. If the lift off force of any one tendon in the total sample population is out of the predicted bounds (less than minimum or greater than maximum), an adjacent tendon on each side of the defective tendon shall also be checked for lift off force. If both of these tendons are found acceptable, the surveillance program may proceed considering the single deficiency as unique and acceptable. More than one defective tendon out of the original sample population is evidence of abnormal degradation of the containment structure. Unless there is evidence of abnormal degradation of the containment structure during the first three tests of the tendons, the number of tendons checked for lift off force during subsequent tests may be reduced to a representative sample of at least 9 tendons (3 dome, 3 vertical and 3 hoop).

SURVEILLANCE REQUIREMENTS (Continued)

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

1. The tendon wires or strands are free of corrosion.

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

3. A minimum tensile strength value of 14,109 psi (guaranteed ultimate strength of the tendon material) for at least three wire or strand samples (one from each end and one at mid-length) cut from each removed wire or strand. Failure of any one of the tendon samples to meet the minimum tensile strength test is evidence of abnormal degradation of the containment structure.

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

4.6.1.6.3 Liner Plate The structural integrity of the containment liner plate shall be determined during the shutdown for each Type A containment leakage rate test (reference Specification 4.6.1.2) by a visual inspection of the plate and verifying no apparent changes in appearance or other abnormal degradation.

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

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 BWST on a containment spray actuation signal and manually transferring suction to the containment sump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.1 Each containment spray system shall be demonstrated OPERABLE:

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

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

   1. Verifying that each automatic valve in the flow path actuates to its correct position on a containment spray actuation test signal.

   2. Verifying that each spray pump starts automatically on a containment spray actuation test signal.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months by verifying a total leak rate $\leq$ 6 gallons per hour for the system at:

1. Normal operating pressure or a hydrostatic test pressure of $\geq$ 190 psig for those parts of the system downstream of the pump suction isolation valve, and

2. $\geq$ 55 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

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

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with spray additive tanks containing at least:

a. A contained volume of between 11,190 and 12,010 gallons of solution containing between 212,000 and 223,000 ppm of sodium hydroxide (NaOH), and

b. A contained volume of between 12,970 and 13,920 gallons of solution containing between 257,000 and 310,000 ppm of sodium thiosulfate ($Na_2S_2O_3$), between 1640 and 2452 ppm of boron, and between 5500 and 7500 ppm of sodium hydroxide (NaOH).

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.2.2 The spray additive system shall be demonstrated OPERABLE:

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

b. At least once per 6 months by:

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

   2. Verifying the concentration of the NaOH and $Na_2S_2O_3$ solutions by chemical analysis.

SURVEILLANCE REQUIREMENTS (Continued)

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

d. At least once per 5 years by verifying each solution flow rate from the following drain connections in the spray additive system:

1.	BSV-101	24.6 ± 3 gpm
2.	BSV-102	17.6 ± 2 gpm
3.	BSV-103	23.3 ± 3 gpm
4.	BSV-104	24.6 ± 3 gpm

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.3 At least two independent containment cooling units shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one of the above required containment cooling units inoperable, restore at least two units to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.3 At least the above required cooling units shall be demonstrated OPERABLE:

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

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

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

   3. Verifying a cooling water flow rate of $\geq$ 500 gpm to each unit cooler.

b. At least once per 18 months by verifying that each unit starts automatically on low speed upon receipt of a containment cooling actuation test signal.

3/4.6.3 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.3.1 The containment isolation valves specified in Table 3.6-1 shall be OPERABLE with isolation times as shown in Table 3.6-1.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more of the isolation valve(s) specified in Table 3.6-1 inoperable, 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 3.6-1 shall be demonstrated OPERABLE prior to returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit by performance of a cycling test and verification of isolation time.

SURVEILLANCE REQUIREMENTS (Continued)

4.6.3.1.2 Each isolation valve specified in Table 3.6-1 shall be demonstrated OPERABLE during the COLD SHUTDOWN or REFUELING MODE at least once per 18 months by:

a. Verifying that on a containment isolation test signal, each automatic isolation valve actuates to its isolation position.

b. Verifying that on a containment radiation-high test signal, each purge and exhaust automatic valve actuates to its isolation position.

TABLE 3.6-1

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
A.	CONTAINMENT ISOLATION		
1.	BSV-27 check #	iso. dur. nor. operation	NA
	BSV-3 #	and open dur. RB spray	60
	BSV-26 check #		NA
	BSV-4 #		60
2.	CAV-126	iso. CA sys. fr. RC letdr.	60
	CAV-1	iso. CA sys. fr. pzr.	60
	CAV-3		60
	CAV-2	iso. CA sys.	60
	CAV-4 #	isolate liquid sampling	60
	CAV-6 #	system	60
	CAV-5 #		60
	CAV-7 #		60
3.	CFV-20 check	iso. N_2 supply fr. CFT-1A	NA
	CFV-28		60
	CFV-17 check	iso. N_2 supply fr. CFT-1B	NA
	CFV-27		60
	CFV-18 check	iso. MU system fr. CFT-1B	NA
	CFV-26		60
	CFV-19 check	iso. MU system fr. CFT-1A	NA
	CFV-25		60
	CFV-42	iso. liquid sampling fr. CF system	60
	CFV-15	iso. WD sys. fr. CF tanks	60
	CFV-16		60
	CFV-29		60
	CFV-11	iso. CF tanks fr. liquid	60
	CFV-12	sampling system	60
4.	CIV-41	iso. CI sys. fr. RB	60
	CIV-40		60
	CIV-34		60
	CIV-35		60

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

	VALVE NUMBER	FUNCTION	ISOLATION TIME (seconds)
5.	DHV-93 check	iso. DH system fr. pzr.	NA
	CHV-91		60
	DHV-43 #	iso. DH sys. fr. RB sump	120
	DHV-42 #		120
	DHV-4# & 41#	iso. DH sys. fr. RC	120
	DHV-6 #	iso. DH system from	60
	DHV-5 #	Reactor Vessel	60
6.	DWV-162 check	iso. system	NA
	DWV-160		60
7.	FWV-44 check #	iso. feedwater	NA
	FWV-45 check #	from RCSG-1A	NA
	FWV-43 check #	iso. feedwater	NA
	FWV-45 check #	from RCSG-1B	NA
8.	MSV-130 #	from RCSG-1A	60
	MSV-148 #	from RCSG-1B	60
	MSV-411 #	iso. main steam lines from RCSG-1A	60
	MSV-412 #	iso. main steam lines from RCSG-1A	60
	MSV-413 #	iso. main steam lines from RCSG-1B	60
	MSV-414 #	iso. main steam lines from RCSG-1B	60
9.	MUV-40	iso. MU system from RC	60
	MUV-41		60
	MUV-49		60
	MUV-253		60
	MUV-261	iso. MU system from	60
	MUV-260	control bleed-off	60
	MUV-259		60
	MUV-258		50

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME (seconds)
9.	(Continued)		
	MUV-162 check	iso. MU system from	NA
	MUV-18	RC pump seal	60
	MUV-163 check #	open during HPI and	NA
	MUV-25 #	iso. dur. nor. operation	60
	MUV-164 check #		NA
	MUV-26 #		60
	MUV-160 check #	open during HPI and	NA
	MUV-23 #	iso. dur. nor. operation	60
	MUV-161 check #	open during HPI and	NA
	MUV-24 #	iso. dur. nor. operation	60
	MUV-27 #	open dur. nor. operation and closed during HPI	60
10.	SWV-39 #	iso. NSCCC from AHF-1C	60
	SWV-45 #		60
	SWV-35 #	iso. NSCCC from AHF-1A	60
	SWV-41 #		60
	SWV-37 #	iso. NSCCC from AHF-1B	60
	SWV-43 #		60
	SWV-48 #	to isolate NSCCC from	60
	SWV-47 #	MUHE-1A & 1B and WDT-5	60
	SWV-49 #		60
	SWV-50 #		60
	SWV-80 #	iso. NSCCC from RCP-1A	60
	SWV-84 #		60
	SWV-82 #	iso. NSCCC from RCP-1C	60
	SWV-86 #		60
	SWV-81 #	iso. NSCCC from RCP-1D	60
	SWV-85 #		60
	SWV-79 #	iso. NSCCC from RCP-1B	60
	SWV-83 #		60
	SWV-109#	NSCCC to DRRD-1	60
	SWV-110#		60

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME (seconds)
11.	WDV-4	iso. WDT-4 from RB sump	60
	WDV-3		60
	WDV-60 & 61	iso. WDT-4 from WDT-5	60
	WDV-94 & 62	iso. WDT-4 from WDP-8	60
	WDV-406	iso. gas waste disposal	60
	WDV-405	from vents in RC system	60
12.	WSV-3	iso. containment monitoring	60
	WSV-4	system from RB	60
	WSV-5		60
	WSV-6		60
B.	CONTAINMENT PURGE AND EXHAUST		
1.	AHV-1C & 1D	iso. pur. sup. system	60
	AHV-1B & 1A	iso. pur. exhaust system	60
C.	MANUAL		
1.	IAV-28	iso. IA from RB	NA
	IAV-29		NA
2.	LRV-50	iso. leak rate test system	NA
	LRV-36	from RB	NA
	LRV-51	iso. atmos. vent and RB	NA
	LRV-35 & 47	purge exhaust system from RB	NA
	LRV-49	iso. atmos. vent from RB	NA
	LRV-38 & 52		NA
	LRV-45	iso. LR test panel from RB	NA
	LRV-44		NA
3.	MSV-146#	iso. misc. waste storage tank from RCS [illegible] RB	NA
4.	NGV-62	iso. NG system from	NA
	NGV-81 #	steam generators	NA
	NGV-82	iso. NG system from pzr.	NA

TABLE 3.6-1 (Continued)

CONTAINMENT ISOLATION VALVES

VALVE NUMBER		FUNCTION	ISOLATION TIME (seconds)
5.	SAV-24	iso. SA from RB	NA
6.	SFV-18	iso. SF system	NA
	SFV-19		NA
	SFV-119#	iso. Fuel Transfer tubes	NA
	SFV-120#	from F.T. Canal	NA
7.	WSV-1	containment monitoring	NA
	WSV-2	system from RB	NA
D.	PENETRATIONS REQUIRING TYPE B TESTS		
	Blind Flange 119	iso. RB	NA
	Blind Flange 120		NA
	Blind Flange 116		NA
	Blind Flange 202		NA
	Blind Flange 348	iso. fuel transfer tube from	NA
	Blind Flange 436	Transfer Canal	NA
	Equipment Hatch	iso. RB	NA
	Personnel Hatch	iso. RB	NA

Not subject to Type C Leakage Test

20

3/4.6.4 COMBUSTIBLE GAS CONTROL

HYDROGEN ANALYZERS

LIMITING CONDITION FOR OPERATION

3.6.4.1 A containment hydrogen analyzer and a gas chromatograph shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.6.4.1 Each hydrogen analysis device shall be demonstrated OPERABLE at least once per 92 days on a STAGGERED TEST BASIS by performing a CHANNEL CALIBRATION using sample gases containing:

a. One volume percent hydrogen, balance nitrogen, and

b. Four volume percent hydrogen, balance nitrogen.

HYDROGEN PURGE SYSEM

LIMITING CONDITION FOR OPERATION

3.6.4.2 A containment hydrogen purge system shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTION:

With the containment hydrogen purge system inoperable, restore the hydrogen purge 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 The hydrogen purge system shall be demonstrated OPERABLE:

a. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

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 purge system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c* and C.5.d* of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 50,000 cfm $\pm$ 10%.

   2. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

*The air flow distribution test of Section 8 of ANSI N510-1975 may be performed downstream of the HEPA filters.

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying a system flow rate of 50,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 50,000 cfm ± 10%.

e. 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 system at a flow rate of 25,000 cfm ± 10%.

f. 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 system at a flow rate of 25,000 cfm ± 10%.

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 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 Nuclear Overpower 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. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional Surveillance Requirements other than those required by Specification 4.0.5, are applicable for the main steam line code safety valves of Table 4.7-1.

TABLE 3.7-1

MAXIMUM ALLOWABLE NUCLEAR OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Steam Generator	Maximum Allowable Nuclear Overpower Trip Setpoint (Percent of RATED THERMAL POWER)
1	85
2	70
3	55

TABLE 4.7-1

STEAM LINE SAFETY VALVES

VALVE NUMBER	LIFT SETTING (± 1%) (psig)	ORIFICE SIZE (inches)
STEAM GENERATOR 3A		
Main steam line A1		
MSV - 34	1050	4.515
MSV - 38	1070	4.515
MSV - 43	1090	4.515
MSV - 40	1100	3.750
Main steam line A2		
MSV - 33	1050	4.515
MSV - 37	1070	4.515
MSV - 42	1090	4.515
MSV - 46	1100	4.515
STEAM GENERATOR 3B		
Main steam line B1		
MSV - 35	1050	4.515
MSV - 39	1070	4.515
MSV - 44	1090	4.515
MSV - 47	1100	4.515
Main steam line B2		
MSV - 36	1050	4.515
MSV - 41	1070	4.515
MSV - 45	1090	4.515
MSV - 48	1100	3.750

PLANT SYSTEMS

EMERGENCY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 Two independent steam generator emergency feedwater pumps and associated flow paths shall be OPERABLE with:

a. One emergency feedwater pump capable of being powered from an OPERABLE emergency bus, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one emergency feedwater system inoperable, restore the inoperable system to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.2 Each emergency feedwater system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

1. Verifying that each steam turbine driven pump develops a discharge pressure of $\geq$ 1100 psig on recirculation flow when the secondary steam supply pressure is greater than 200 psig.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on an emergency feedwater actuation test signal.

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

3. Verifying that the operating air accumulators for FWV-39 and FWV-40 maintain $\geq$ 27 psig for at least one hour when isolated from their air supply.

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tank (CST) shall be OPERABLE with a minimum contained volume of 150,000 gallons of water.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

a. Restore the CST to OPERABLE status or be in HOT SHUTDOWN within the next 12 hours, or

b. Demonstrate the OPERABILITY of the condenser hotwell as a backup supply to the emergency feedwater system and restore the condensate storage tank to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

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

4.7.1.3.2 The condenser hotwell shall be demonstrated OPERABLE at least once per 12 hours by verifying a minimum contained volume of 150,000 gallons of water whenever the condenser hotwell is the supply source for the emergency feedwater system.

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	SAMPLE AND ANALYSIS FREQUENCY
1. Gross Beta Activity Determination	At least once per 72 hours
2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	a) 1 per 31 days, whenever the gross Beta activity determination indicates iodine concentrations greater than 10% of the allowable limit. b) 1 per 6 months, whenever the gross Beta 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 provided:

a. The inoperable isolation valve is maintained closed.

Otherwise, be in HOT SHUTDOWN within the next 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.5 Each main steam line isolation valve shall be demonstrated OPERABLE by verifying full closure within 5 seconds when tested pursuant to Specification 4.0.5.

PLANT SYSTEMS

SECONDARY WATER CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.7.1.6 The secondary water chemistry shall be maintained within the limits of Table 3.7-2.

APPLICABLITY: MODES 1, 2 and 3.

ACTION:

(To be determined in a manner set forth in the bases 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 3.7-2

SECONDARY WATER CHEMISTRY LIMITS

Water Sample Location	Parameters*
*	*

*Sample locations, parameters and limits to be established in approximatley 6 months following issuance of the full power license 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 following issuance of the full power license based upon test program described in bases.

PLANT SYSTEMS

3/4.7.2 STEAM GENERATOR PRESSURE/TEMPERATURE LIMITATION

LIMITING CONDITION FOR OPERATION

3.7.2.1 The temperature of the secondary coolant in the steam generators shall be > 110°F when the pressure of the secondary coolant in the steam generator is > 237 psig.

APPLICABILITY: At all times.

ACTION:

With the requirements of the above specification not satisfied:

a. Reduce the steam generator pressure to ≤ 237 psig within 30 minutes, and

b. Perform an engineering evaluation to determine the effect of overpressurization on the structural integrity of the steam generator. Determine that the steam generator remains acceptable for continued operation prior to increasing its pressure above 237 psig.

SURVEILLANCE REQUIREMENTS

4.7.2.1 The temperature of the secondary coolant in each steam generator shall be determined to be > 110°F at least once per hour when secondary pressure in the steam generator is > 237 psig and T_{avg} is < 200°F.

3/4.7.3 CLOSED CYCLE COOLING WATER SYSTEMS

NUCLEAR SERVICES CLOSED CYCLE COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.1 The nuclear services closed cycle cooling system shall be OPERABLE and shall consist of a minimum of:

a. Two emergency nuclear services closed cycle cooling water pumps, each powered from a separate OPERABLE emergency bus, and

b. Three nuclear services heat exchangers.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one emergency nuclear service closed cycle cooling water pump OPERABLE or with only two nuclear services heat exchangers OPERABLE, restore a minimum of two pumps and three heat exchangers 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 The nuclear services closed cycle cooling system shall be demonstrated OPERABLE:

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

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

1. Verifying that each automatic valve in the flow path actuates to its correct position on an ESAS test signal.

2. Verifying that each emergency nuclear services closed cycle cooling water pump starts automatically upon receipt of an ESAS test signal.

DECAY HEAT CLOSED CYCLE COOLING WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.3.2 Two independent decay heat closed cycle cooling water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one decay heat closed cycle cooling water loop OPERABLE, restore the inoperable loop 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

4.7.3.2 Each decay heat closed cycle cooling water loop shall be demonstrated OPERABLE:

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

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

   1. Verifying that each automatic valve in the flow path actuates to its correct position on an ESAS test signal.

   2. Verifying that each decay heat closed cycle cooling water pump starts automatically upon receipt of an ESAS test signal.

3/4.7.4 SEA WATER SYSTEMS

NUCLEAR SERVICES SEA WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.1 At least two independent emergency nuclear services sea water pumps and the associated flow path shall be OPERABLE with each pump capable of being powered from separate OPERABLE emergency busses.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one emergency nuclear services sea water pump OPERABLE, restore the inoperable 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

4.7.4.1 The nuclear service sea water systems shall be demonstrated OPERABLE:

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

b. At least once per 18 months, during shutdown, by verifying that each emergency nuclear services sea water pump starts automatically upon receipt of an ESAS test signal.

PLANT SYSTEMS

DECAY HEAT SEA WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4.2 Two independent decay heat sea water loops shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With only one decay heat sea water loop OPERABLE, restore the inoperable loop 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

4.7.4.2 Each decay heat sea water loop shall be demonstrated OPERABLE:

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

b. At least once per 18 months, during shutdown, by verifying that each decay heat sea water pump starts automatically upon receipt of an ESAS test 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. A minimum water level at or above elevation 81 feet Plant Datum,

b. An inlet water temperature of $\leq$ 105°F, and

c. A maximum elevation of the intake canal bottom of 74 feet Plant Datum.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With the water level < 81 feet Plant Datum or the inlet water temperature > 105°F, be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the elevation of the intake canal bottom > 74 feet Plant Datum, restore the elevation of the intake canal bottom to $\leq$ 74 feet Plant Datum within 90 days or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.5.1 The ultimate heat sink shall be determined OPERABLE:

a. At least once per 24 hours by verifying the inlet water temperature and water level to be within their limits, and

b. At least once per 24 months by determining that the maximum elevation of the intake canal bottom is $\leq$ 74 feet Plant Datum.

3/4.7.6 FLOOD PROTECTION

LIMITING CONDITION FOR OPERATION

3.7.6.1 Flood protection shall be provided for all safety related systems, components and structures when:

a. The water level of the Gulf of Mexico exceeds 98 feet Plant Datum, at the intake structure, and

b. A Hurricane Warning is in effect.

APPLICABILITY: At all times.

ACTION:

With the water level at the intake structure above elevation 98 feet Plant Datum and with a Hurricane Warning in effect:

a. Close all watertight doors providing flood protection for safety related systems, components and structures within 2 hours, and

b. Be in at least HOT SHUTDOWN within 6 hours; be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.6.1.1 When a Hurricane Watch or Warning is in effect, the water level at the intake structure shall be determined to be within the limit by:

a. Measurement at least once per 12 hours when the water level is below elevation 92 feet Plant Datum,

b. Measurement at least once per 4 hours when the water level is equal to or above elevation 92 feet Plant Datum, and

c. Measurement at least once per 30 minutes when the water level is equal to or above elevation 94 feet Plant Datum.

4.7.6.1.2 Meteorological forecasts shall be obtained from the National Hurricane Center in Miami, Florida at least once per 2 hours when a Hurricane Warning is in effect.

3/4.7.7 CONTROL ROOM EMERGENCY VENTILATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.7.1 Two independent control room emergency ventilation systems shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.7.7.1 Each 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 $\leq$ 120°F.

b. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

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

*The air flow distribution test of Section 8 of ANSI N510-1975 may be performed downstream of the HEPA filters.

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that with the system operating at a flow rate of 43,500 cfm ± 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is ≤ 1% when the system is tested by admitting cold DOP at the system intake.

2. Verifying that the ventilation system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c* and C.5.d* of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 43,500 cfm ± 10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

4. Verifying a system flow rate of 43,500 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 43,500 cfm ± 10%.

2. Verifying that on a ESFAS test signal and a high radiation test signal, the system automatically switches into a recirculation mode of operation with flow through the HEPA filters and charcoal adsorber banks.

SURVEILLANCE REQUIREMENTS (Continued)

f. 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 system at a flow rate of 43,500 cfm $\pm$ 10%.

g. 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 system at a flow rate of 43,500 cfm $\pm$ 10%.

3/4.7.8 AUXILIARY BUILDING VENTILATION EXHAUST SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.8.1 The auxiliary building ventilati·ı exhaust system shall be OPERABLE and shall consist of a minimum o‹ two independent pairs of exhaust fans and four filter systems.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one pair of exhaust fans or one filter system inoperable, restore the inoperable pair of fans or system to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.7.8.1 Each auxiliary building ventilaiton exhaust system shall be demonstrated OPERABLE:

a. At least once per 31 days on a STAGGERED TEST BASIS by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes.

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

   1. Verifying that with the system operating at a flow rate of 156,680 cfm $\pm$ 10% and exhausting through the HEPA filters and charcoal adsorbers, the total bypass flow of the system to the facility vent, including leakage through the system diverting valves, is $\leq$ 1% when the system is tested by admitting cold DOP at the system intake.

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*The air flow distribution test Section 8 of ANSI N510-1975 may be performed downstream of the HEPA filters.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying that the ventilation system satisfies the in-place testing acceptance criteria and uses the test procedures of Regulatory Positions C.5.a, C.5.c* and C.5.d* of Regulatory Guide 1.52, Revision 1, July 1976, and the system flow rate is 156,680 cfm ±10%.

3. Verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

4. Verifying a system flow rate of 156,680 cfm ±10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 1, July 1976, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 1, July 1976.

d. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 156,680 cfm ± 10%.

e. 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 system at a flow rate of 39,170 cfm ± 10%.

f. 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 system at a flow rate of 39,170 cfm ± 10%.

3/4.7.9 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

3.7.9.1 All hydraulic snubbers listed in Table 3.7-3 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic snubbers inoperable, replace or 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

4.7.9.1 Hydraulic snubbers will be demonstrated OPERABLE by performance of the following augmented inservice inspection program and the requirements of Specification 4.0.5.

a. 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 4.7-4 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 4.7-4 inspection schedule shall be made assuming the unit was previously at the 6 month inspection interval.

b. 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.

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-3, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lbs capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified in Table 3.7-3 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 3.7-3

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)
	Core Flood System			
CFH-12	116'-7"	A	No	Yes
CFH-13	111'-4"	A	No	No
CFH-14	110'-6"	A	No	No
CFH-15	120'-9"	I	Yes	Yes
CFH-16	120'-9"	I	Yes	No
CFH-17	120'-9"	I	Yes	No
CFH-18	120'-9"	I	Yes	No
CFH-19	123'-0"	I	Yes	Yes
	Decay Heat Removal System			
DHH-17	110'-6"	A	No	No
DHH-18	110'-6"	A	No	No
DHH-19	109'-0"	A	No	No
DHH-20	115'-0"	I	Yes	No
DHH-21	117'-0"	I	Yes	Yes
DHH-22	117'-0"	I	Yes	Yes
DHH-23	110'-6"	A	No	Yes
DHH-24	109'-3"	A	No	No
DHH-25	110'-6"	A	No	Yes
DHH-26H	110'-6"	A	No	No
DHH-26V	110'-6"	A	No	No
DHH-27	110'-6"	A	No	Yes

TABLE 3.7-3

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)
	Emergency Feedwater System			
EFH-14U	115'-0"	A	No	No
EFH-14L	115'-0"	A	No	No
EFH-15U	115'-0"	A	No	No
EFH-15L	115'-0"	A	No	No
EFH-27	145'-9"	I	Yes	Yes
EFH-28	145'-9"	I	Yes	Yes
EFH-92	140'-0"***	A	No	No
EFH-93	131'-8"***	A	No	No
EFH-94	131'-8"***	A	No	No
EFH-95	140'-0"***	A	No	No
EFH-96	141'-3"***	A	No	No
EFH-106	141'-3"***	A	No	No
EFH-107	141'-3"***	A	No	No
EFH-108	141'-3"***	A	No	No
EFH-109	133'-0"***	A	No	No
EFH-110	133'-0"***	A	No	No
EFH-141	126'-6"***	A	No	No
EFH-143	133'-6"***	A	No	No
EFH-144	141'-3"***	A	No	No
	Make-up and Purification System			
MUH-32	115'-11"	I	Yes	No
MUH-33	110'-6"	A	No	Yes
MUH-34	110'-6"	A	No	No

TABLE 3.7-3

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)
	Make-up and Purification System (Continued)			
MUH-35	110'-6"	I	Yes	No
MUH-36	109'-0"	A	No	Yes
MUH-37	109'-0"	A	No	No
MUH-38	123'-10"	I	Yes	Yes
MUH-39	112'-0"	I	Yes	No
MUH-40	112'-0"	I	Yes	No
MUH-41	129'-1"	I	Yes	No
MUH-42	119'-0"	I	Yes	No
MUH-43	114'-0"	I	Yes	No
MUH-44	114'-0"	A	No	No
MUH-45	114'-0"	A	No	No
MUH-46	114'-0"	I	Yes	No
MUH-47	114'-0"	A	No	Yes
MUH-48	114'-0"	A	No	Yes
MUH-49	108'-6"	A	No	Yes
MUH-50	129'-9"	I	Yes	Yes
MUH-51	117'-0"	I	Yes	Yes
MUH-52	110'-0"	A	No	No
MUH-53	110'-0"	A	No	No
MUH-80	108'-0"	I	Yes	Yes
MUH-81	108'-7"	A	No	No
MUH-82	108'-7"	A	No	No
MUH-83	108'-7"	A	No	No
MUH-84	108'-7"	A	No	No
MUH-85	104'-0"	A	No	No

TABLE 3.7-3

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 Nc)
	Make-up and Purification System (Continued)			
MUH-86	102'-3"	A	No	No
MUH-87	102'-3"	A	No	No
MUH-88	103'-0"	I	Yes	No
MUH-89	108'-0"	A	No	No
	Decay Heat Removal System			
DHH-35	152'-5"	I	Yes	No
DHH-36	152'-5"	I	Yes	No
DHH-37	159'-7"	I	Yes	Yes
DHH-38	160'-1"	I	Yes	Yes
DHH-39	165'-9"	I	Yes	No
DHH-661	86'-6"***	I	Yes	No
DHR-18	84'-7"***	I	Yes	No
DHR-21	103'-6"***	A	No	No
DHR-24U	129'-6"***	A	No	No
DHR-24L	129'-6"***	A	No	No
DHR-28	134'-4"***	A	No	No
DHR-31	84'-9"***	I	Yes	No
DHR-37	85'-6"***	I	Yes	No
DHR-49	85'-6"***	I	Yes	No

TABLE 3.7-3

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)
	Reactor Coolant System			
RCH-26	171'-1"	I	Yes	No
RCH-27	171'-1"	I	Yes	No
RCH-28	168'-8"	I	Yes	No
RCH-29	168'-8"	I	Yes	No
RCH-30	168'-8"	I	Yes	No
RCH-31	168'-8"	I	Yes	No
RCH-32	159'-6"	I	Yes	No
RCH-33	159'-6"	I	Yes	No
RCH-34	147'-0"	A	No	No
RCH-35	142'-6"	A	No	Yes
RCH-36	144'-9"	A	No	Yes
RCH-37	144'-9"	A	No	No
RCH-38	144'-9"	A	No	No
RCH-39	147'-0"	A	No	Yes
RCH-40	147'-0"	A	No	Yes
RCH-47N	121'-9"	I	Yes	Yes
RCH-47S	121'-9"	I	Yes	Yes
RCH-48	121'-9"	I	Yes	Yes
RCH-49	121'-9"	I	Yes	Yes
RCH-50	167'-5"	I	Yes	No
RCH-51	159'-6"	I	Yes	Yes
RCH-52	147'-0"	A	No	No
RCH-53	147'-0"	A	No	No
RCH-54	147'-0"	I	Yes	No
RCH-55	123'-9"	A	No	No
RCH-56	158'-6"	I	Yes	No
RCH-57	158'-3"	I	Yes	Yes
RCH-58	120'-0"	A	No	No
RCH-59	168'-0"	I	Yes	No

TABLE 3.7-3

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)
	Reactor Coolant System (Continued)			
RCH-60	167'-5"	I	Yes	No
RCH-61	145'-6"	A	No	No
RCH-62	148'-4"	I	Yes	Yes
RCH-63	148'-4"	I	Yes	No
RCH-64	148'-3"	I	Yes	Yes
RCH-65	131'-6"	I	Yes	Yes
RCH-66	140'-0"	I	Yes	No
RCH-67	166'-0"	I	Yes	No
RCH-68	167'-1"	I	Yes	No
RCH-69	167'-1"	I	Yes	No
RCH-70	167'-1"	I	Yes	No
RCH-71U	167'-1"	I	Yes	No
RCH-71L	167'-1"	I	Yes	No
RCH-73	167'-1"	I	Yes	No
RCH-74	167'-1"	I	Yes	No
RCH-76	139'-11"	I	Yes	Yes
RCH-77	131'-6"	I	Yes	Yes
RCH-78	150'-10"	I	Yes	Yes
RCH-79	156'-8"	I	Yes	Yes
RCH-80	168'-3"	I	Yes	No
RCH-81	168'-4"	I	Yes	No
RCH-82E	143'-9"	A	No	Yes
RCH-82W	143'-9"	A	No	Yes
RCH-83	145'-11"	A	No	No
RCH-84	123'-9"	A	No	No
RCH-85	147'-0"	A	No	No
RCH-86	144'-9"	A	No	No
RCH-87	147'-0"	A	No	No
RCH-88	147'-0"	A	No	No
RCH-89	124'-0"	A	No	No
RCH-90	120'-0"	A	No	No

TABLE 3.7-3

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)
	Reactor Coolant Pump 3A1			
RCHS-1A	134'-6"	I	Yes	Yes
RCHS-2A	134'-6"	I	Yes	Yes
RCHS-3A	134'-6"	I	Yes	Yes
RCHS-4A	134'-6"	I	Yes	Yes
RCHS-5A	134'-6"	I	Yes	Yes
RCHS-6A	134'-6"	I	Yes	Yes
RCHS-7A	134'-6"	I	Yes	Yes
RCHS-8A	134'-6"	I	Yes	Yes
	Reactor Coolant Pump 3A2			
RCHS-1B	134'-6"	I	Yes	Yes
RCHS-2B	134'-6"	I	Yes	Yes
RCHS-3B	134'-6"	I	Yes	Yes
RCHS-4B	134'-6"	I	Yes	Yes
RCHS-5B	134'-6"	I	Yes	Yes
RCHS-6B	134'-6"	I	Yes	Yes
RCHS-7B	134'-6"	I	Yes	Yes
RCHS-8B	134'-6"	I	Yes	Yes
	Reactor Coolant Pump 3B1			
RCHS-1C	134'-6"	I	Yes	Yes
RCHS-2C	134'-6"	I	Yes	Yes
RCHS-3C	134'-6"	I	Yes	Yes
RCHS-4C	134'-6"	I	Yes	Yes
RCHS-5C	134'-6"	I	Yes	Yes
RCHS-6C	134'-6"	I	Yes	Yes
RCHS-7C	134'-6"	I	Yes	Yes
RCHS-8C	134'-6"	I	Yes	Yes

TABLE 3.7-3

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)
	Reactor Coolant Pump 3B2			
RCHS-1D	134'-6"	I	Yes	Yes
RCHS-2D	134'-6"	I	Yes	Yes
RCHS-3D	134'-6"	I	Yes	Yes
RCHS-4D	134'-6"	I	Yes	Yes
RCHS-5D	134'-6"	I	Yes	Yes
RCHS-6D	134'-6"	I	Yes	Yes
RCHS-7D	134'-6"	I	Yes	Yes
RCHS-8D	134'-6"	I	Yes	Yes

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

**All safety related snubbers are located in the Reactor Building except those with the triple asterisk.

****Modifications to this table due to changes in high radiation areas shall be submitted to the NRC as part of the next License amendment request.

TABLE 4.7-4

HYDRAULIC SNUBBER INSPECTION SCHEDULE

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

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10.1 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

4.7.10.1.1 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.

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

a. Sources in use (excluding startup sources and fission detectors 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 and fission detector shall be tested prior to use or transfer to another licensee unless tested within the previous six months. Sealed sources and fission detectors transferred without a certificate indicating the last test date shall be tested prior to being placed into use.

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

4.7.10.1.3 Reports - A report shall be prepared and submitted to the Commission on an annual basis if sealed source or fission detector leakage tests reveal the presence of $\geq$ 0.005 microcuries of removable contamination.

3/4.8 ELECTRICAL POWER SYSTEMS

3/4.8.1 A.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.1.1 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

b. Two separate and independent diesel generators each with:

1. A separate day fuel tank containing a minimum volume of 400 gallons of fuel,

2. A separate fuel storage system containing a minimum volume of 20,300 gallons of fuel, and

3. A separate fuel transfer pump.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

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

ACTION (Continued)

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

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

SURVEILLANCE REQUIREMENTS

4.8.1.1.1 Each independent circuit between the offsite transmission network and the onsite Class 1E distribution system shall be:

a. Determined OPERABLE at least once per 7 days by verifying;

1. Correct breaker alignments and indicated power availability, and

2. That the sump pumps in the tunnel containing the DC control feeds to the 230kv switchgear are OPERABLE.

b. Demonstrated OPERABLE at least once per 18 months during shutdown by transferring unit power supply from the normal circuit to the alternate circuit.

SURVEILLANCE REQUIREMENTS

c. Demonstrated OPERABLE by determining that each battery supplying DC control power to the 230kv switchyard breakers is OPERABLE;

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

   a) The electrolyte level of each pilot cell is between the minimum and maximum level indication marks,

   b) The pilot cell specific gravity, corrected to 77°F, and full electrolyte level is $\geq$ 1.20.

   c) The pilot cell voltage is $\geq$ 2.15 volts, and

   d) The overall battery voltage is $\geq$ 120 volts.

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

   a) The voltage of each connected cell is $\geq$ 2.15 volts under float charge and has not decreased more than 0.10 volts from the value observed during the baseline tests, and

   b) The specific gravity, corrected to 77°F, and full electrolyte level of each connected cell is $\geq$ 1.20 and has not decreased more than 0.01 from the value observed during the previous tests, and

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

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

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

   b) The cell-to-cell and terminal connections are clean, tight and coated with anti-corrosion materials,

   c) The battery charger will supply at least 95 amperes at 125 volts for at least 2 hours.

SURVEILLANCE REQUIREMENTS (Continued)

4. At least once per 18 months, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 1 hour when the battery is subjected to a battery service test.

5. At least once per 60 months, 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.

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE:

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

1. Verifying the fuel level in the day fuel tank,
2. Verifying the fuel level in the fuel storage tank,
3. Verifying the fuel transfer pump can be started and transfers fuel from the storage system to the day tank,
4. Verifying the diesel starts from ambient condition and accelerates to at least 900 rpm in $\leq$ 10 seconds,
5. Verifying the generator is synchronized, loaded to $\geq$ 1500 kw, and operates for $\geq$ 60 minutes, and
6. Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

b. At least once each 92 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 diesel to an inspection in accordance with procedures prepared in conjunction with its manufacturer's recommendations for this class of standby service,

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying the generator capability to reject a load of $\geq$ 515 kw without tripping,

3. Simulating a loss of offsite power in conjunction with reactor building high pressure and reactor building high-high pressure test signals, and;

   a) Verifying de-energization of the emergency buses and load shedding from the emergency busses,

   b) Verifying that the 4160 v. emergency bus tie breakers open.

   c) Verifying the diesel starts from ambient condition on the auto-start signal, energizes the emergency busses with permanently connected loads, energizes the auto-connected emergency loads through the load sequencer and operates for $\geq$ 5 minutes while its generator is loaded with the emergency loads.

4. Verifying the diesel generator operates for $\geq$ 60 minutes while loaded to $\geq$ 3000 kw,

5. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 3000 kw, and

6. Verifying that the automatic load sequence timers are OPERABLE with each load sequence time within $\pm$ 10%.

ELECTRICAL POWER SYSTEMS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. One diesel generator with:

1. Day fuel tank containing a minimum volume of 400 gallons of fuel,

2. A fuel storage system containing a minimum volume of 20,300 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 performance of each of the Surveillance Requirements of 4.8.1.1.1 and 4.8.1.1.2, except requirement 4.8.1.1.2.a.5.

ELECTRICAL POWER SYSTEMS

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 normal sources of power:

4160 volt Emergency Bus # 3A

4160 volt Emergency Bus # 3B

480 volt Emergency Bus # 3A

480 volt Emergency Bus # 3B

120 volt A.C. Vital Bus # 3A

120 volt A.C. Vital Bus # 3B

120 volt A.C. Vital Bus # 3C

120 volt A.C. Vital Bus # 3D

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

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

b. With transformers, supplied from a single ES train, powering one or two vital busses operation may continue and may be initiated provided the remaining vital busses are powered from inverters.

SURVEILLANCE REQUIREMENTS

4.8.2.1.1 The specified A.C. busses shall be determined OPERABLE and energized from normal A.C. sources at least once per 7 days by verifying correct breaker alignment and indicated power availability.

SURVEILLANCE REQUIREMENTS (Continued)

4.8.2.1.2 The transformers capable of powering vital busses shall be demonstrated OPERABLE at least once per 18 months and within 24 hours after entering ACTION b. above.

ELECTRICAL POWER SYSTEMS

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

1 - 4160 volt Emergency Bus

1 - 480 volt Emergency Bus

2 - 120 volt A.C. Vital Busses

APPLICABILITY: MODES 5 and 6.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.8.2.2 The specified A.C. busses shall be determined OPERABLE and energized from normal A.C. sources at least once per 7 days by verifying correct breaker alignment and 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 250/125-volt D.C. bus No. 3A, 250/125-volt D.C. battery bank No. 3A and two 50% capacity chargers.

TRAIN "B" consisting of 250/125-volt D.C. bus No. 3B, 250/125-volt D.C. battery bank No. 3B, and two 50% capacity chargers.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one 250/125-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 one 250/125-volt D.C. battery and/or a 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 correct breaker alignment and indicated power availability.

4.8.2.3.2 Each 250/125-volt battery bank and charger shall be demonstrated OPERABLE:

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

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

SURVEILLANCE REQUIREMENTS (Continued)

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

3. The pilot cell voltage is $\geq$ 2.15 volts, and

4. The overall battery voltage is $\geq$ 240/120 volts.

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

1. The voltage of each connected cell is $\geq$ 2.15 volts under float charge and has not decreased more than 0.10 volts from the value observed during the original acceptance test, and

2. The specific gravity, corrected to 77°F, and full electrolyte level of each connected cell is $\geq$ 1.20 and has not decreased more than 0.01 from the value observed during the previous test, and

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 and coated with anti-corrosion material,

3. The battery charger will supply at least 190 amperes at a minimum of 120 volts for at least 8 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for 2 hours when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 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

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 - 250/125-volt D.C. bus, and

1 - 250/125-volt battery bank and chargers 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 250/125-volt D.C. bus shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.4.2 The above required 250/125-volt battery bank and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.3.2.

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the 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 $\geq$ 1925 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 $\geq$ 2700 gpm of 2270 ppm boron solution or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 1925 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 safety or regulating rod in excess of 3 feet from its fully inserted position.

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

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, the following instrumentation shall be OPERABLE:

a. Two source range neutron flux monitors, each with visual indication in the control room and one with audible indication in the control room, and

b. One auxiliary source range neutron flux monitor 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 source range neutron flux monitor shall be demonstrated OPERABLE by performance of :

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

REFUELING OPERATIONS

DECAY TIME

LIMITING CONDITION FOR OPERATION

3.9.3 The reactor shall be subcritical for at least 72 hours.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3 The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

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 our bolts,

b. A minimum of one door in each airlock closed, and

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

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

   2. Be capable of being closed by an OPERABLE automatic containment purge and exhaust 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 purge and exhaust valve within 100 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 isolated condition, or

b. Testing the containment purge and exhaust valves per the applicable portions of Specification 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 of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

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

FUEL HANDLING BRIDGE OPERABILITY

LIMITING CONDITION FOR OPERATION

3.9.6 The fuel handling bridges shall be used for movement of control rods or fuel assemblies and shall be OPERABLE with:

a. A hoist minimum capacity of 3000 pounds, and

b. A hoist overload cutoff limit of 2750* pounds.

APPLICABILITY: During movement of control rods or fuel assemblies within the reactor pressure vessel.

ACTION:

With the requirements for bridge OPERABILITY not satisfied, suspend use of any inoperable bridge from operations involving the movement of control rods or fuel assemblies within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.6 Each fuel handling bridge used for movement of control rods or fuel assemblies within the reactor pressure vessel shall be demonstrated OPERABLE within 100 hours prior to the start of moving control rods or fuel assemblies by performing a hoist load test of at least 3000 pounds and demonstrating an automatic load cutoff when the hoist load exceeds 2750* pounds.

*When handling control rods or fuel assemblies with $\geq$ 23 feet of water over the top of fuel assemblies seated in the reactor core; otherwise, the hoist overload cutoff shall be set at 2880 pounds.

CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2750 pounds, except for movement of the missile shield and pool divider gate as necessary for access to the fuel assemblies, shall be prohibited from travel over fuel assemblies in the storage pool.

APPLICABILITY: With fuel assemblies and water 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

4.9.7.1 Crane interlocks and/or physical stops which prevent crane travel with loads in excess of 2750 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane operation and at least once per 7 days during crane operation.

4.9.7.2 Prior to operating the crane in the cask handling mode, verify that:

a. No fuel assemblies are in the storage pool adjacent to the cask loading area, and

b. The watertight gate between storage pools is inplace and sealed.

COOLANT CIRCULATION

LIMITING CONDITION FOR OPERATION

3.9.8 At least one decay heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

a. With less than one decay heat removal 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 decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS to prevent water turbulence problems.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.8 A decay heat removal loop shall be determined to be operating and circulating reactor coolant at a flow rate of $\geq$ 2700 gpm at least once per 24 hours.

CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.9 The containment purge and exhaust isolation system shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTION:

With the containment purge and exhaust isolation system inoperable, close each of the purge and exhaust 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 purge and exhaust isolation system shall be demonstrated OPERABLE within 100 hours prior to the start of and at least once per 7 days during CORE ALTERATIONS by verifying that containment purge and exhaust isolation occurs on manual initiation and on a high radiation test signal from the containment gaseous activity radiation monitoring instrumentation channels.

WATER LEVEL - REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.10 As a minimum, 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 control rods within the reactor pressure vessel while in MODE 6.

ACTION:

With the requirements of the above specification not satisfied, suspend all operation involving movement of fuel assemblies or control rods within the reactor pressure vessel. The provisions of Specification 3.0.3 are not applicable.

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 during movement of fuel assemblies or control rods.

REFUELING OPERATIONS

STORAGE POOL

LIMITING CONDITION FOR OPERATION

3.9.11 All missile shields and at least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the storage racks.

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

ACTION:

a. With all missile shields not installed over the storage pool:

   1) Immediately install all missile shields upon notification of a Tornado Watch, and

   2) Install all missile shields not required to be removed for in-progress handling of fuel assemblies.

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

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.11.1 All missile shields shall be determined to be installed over the storage pool when irradiated fuel assemblies are in the fuel storage pool:

   a) Immediately upon notification of a Tornado Watch, and

   b) Upon completion of handling fuel assemblies.

4.9.11.2 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.

STORAGE POOL VENTILATION

LIMITING CONDITION FOR OPERATION

3.9.12 The auxiliary building ventilation exhaust system servicing the storage pool area shall be OPERABLE.

APPLICABILITY: Whenever irradiated fuel is in the storage pool.

ACTION:

a. With one pair of exhaust fans or two filter systems inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the OPERABLE pair of exhaust fans are in operation and discharging through at least one system of HEPA filters and charcoal adsorbers.

b. With the auxiliary building exhaust ventilation system servicing the storage pool area inoperable, suspend all operations involving movement of fuel within the storage pool or crane operation with loads over the storage pool until at least one pair of exhaust fans and one filter system is restored to OPERABLE status.

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

SURVEILLANCE REQUIREMENTS

4.9.12 The above required auxiliary building ventilation exhaust system servicing the storage pool area shall be demonstrated OPERABLE per the applicable Surveillance Requirements of 4.7.8.1.

3/4.10 SPECIAL TEST EXCEPTIONS

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.1 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained $\leq$ 85% of RATED THERMAL POWER,

b. The Nuclear Overpower Trip Setpoint is $\leq$ 10% of Rated Thermal Power higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% of RATED THERMAL POWER, and,

c. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in 4.10.1.2 below.

APPLICABILITY: MODE 1.

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, 3.1.3.7, 3.2.1 or 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficiently to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

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

SURVEILLANCE REQUIREMENTS

4.10.1.1 The Nuclear Overpower Trip Setpoint shall be determined to be set within the limits specified within 8 hours prior to the initiation of and at least once per 8 hours during PHYSICS TESTS.

4.10.1.2 The Surveillance Requirements of Specifications 4.2.2 and 4.2.3 shall be performed at least once per two hours during PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.2 The limitations of Specifications 3.1.1.3, 3.1.3.1, 3.1.3.2, 3.1.3.5, 3.1.3.6, and 3.1.3.7 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 on the OPERABLE Nuclear Overpower Channels are set at ≤ 25% of RATED THERMAL POWER.

c. The nuclear instrumentation Source Range and Intermediate Range high startup rate control rod withdrawal inhibit are OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the control rod drive trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be < 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.2.2 Each Source and Intermediate Range and Nuclear Overpower Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

SPECIAL TEST EXCEPTION

NO FLOW TEST

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1 may be suspended during the performance of startup and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Nuclear Overpower channels are set $\leq$ 25% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the control rod drive trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each Nuclear Overpower Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTION

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.4 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided:

a. Reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s), and

b. All axial power shaping rods are withdrawn to at least 35% (indicated position) and OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

a. With any safety or regulating control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion or the axial power shaping rods not within their withdrawal limits, immediately initiate and continue boration at $\geq$ 10 gpm of 12,250 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With all safety or regulating control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 10 gpm of 12,250 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The position of each safety, regulating, and axial power shaping rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.4.2 Each safety or regulating control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

4.10.4.3 The axial power shaping rods shall be demonstrated OPERABLE by moving each axial power shaping rod $\geq$ 6.5% (indicated position) within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

BASES

FOR

LIMITING CONDITIONS FOR OPERATION

AND

SURVEILLANCE REQUIREMENTS

NOTE

The summary statements contained in this section provide the bases for the specifications of Sections 3.0 and 4.0 and are not considered a part of these technical specifications as provided in 10 CFR 50.36.

3/4.0 APPLICABILITY

BASES

The specifications of this section provide the general requirements applicable to each of the Limiting Conditions for Operation and Surveillance Requirements within Section 3/4.

3.0.1 This specification defines the applicability of each specification in terms of defined OPERATIONAL MODES or other specified conditions and is provided to delineate specifically when each specification is applicable.

3.0.2 This specification defines those conditions necessary to constitute compliance with the terms of an individual Limiting Condition for Operation and associated ACTION requirement.

3.0.3 This specification delineates the ACTION to be taken for circumstances not directly provided for in the ACTION statements and whose occurrence would violate the intent of the specification. For example, Specification 3.5.1 calls for each Reactor Coolant System core flooding tank to be OPERABLE and provides explicit ACTION requirements when one tank is inoperable. Under the terms of Specification 3.0.3, if more than one 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 Operations. Under this criteria, equipment, systems or components are assumed to be OPERABLE if the associated surveillance activities have been satisfactorily performed within the specified time interval. Nothing in this provision is to be construed as defining equipment, systems or components OPERABLE, when such items are found or known to be inoperable although still meeting the Surveillance Requirements.

4.0.4 This specification ensures that the surveillance activities associated with a Limiting Condition for Operation have been performed within the specified time interval prior to entry into an OPERATIONAL MODE or other applicable condition. The intent of this provision is to ensure that surveillance activities have been satisfactorily demonstrated on a current basis as required to meet the OPERABILITY requirements of the Limiting Condition for Operation.

Under the terms of this specification, for example, during initial plant startup or following extended plant outages, the applicable surveillance activities must be performed within the stated surveillance interval prior to placing or returning the system or equipment into OPERABLE status.

APPLICABILITY

BASES

4.0.5 This specification ensures that inservice inspection of ASME Code Class 1, 2 and 3 components and inservice testing of ASME Code Class 1, 2 and 3 pumps and valves will be performed in accordance with a periodically updated version of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda as required by 10 CFR 50.55a. Relief from any of the above requirements has been provided in writing by the Commission and is not a part of these technical specification.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 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. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits.

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 0.60% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least 2700 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2700 GPM will circulate an equivalent Reactor Coolant System volume of 12,000 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) makeup or DHR pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE emergency busses.

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 either 4088 gallons of 12,250 ppm borated water from the boric acid storage tanks or 24,421 gallons of 2270 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 415,200 gallons of borated water in the borated water storage 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.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the

3/4.1.2 BORATION SYSTEMS (Continued)

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.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% of $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 165 gallons of 12,250 ppm borated water from the boric acid storage system or 888 gallons of 2270 ppm borated water from the borated water storage tank.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between 7.2 and 11.0 of the solution sprayed within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} $\geq$ 525°F and with reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with 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 limitation on THERMAL POWER based on xenon reactivity is necessary to ensure that power peaking limits are not exceeded even with specified rod insertion limits satisfied.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and during short term transients, (b) maintaining the peak linear power density $\leq$ 18.0 kw/ft during normal operation, and (c) maintaining the peak power density $\leq$ 19.7 kw/ft during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power-imbalance envelope defined in Figures 3.2-1 and 3.2-2, and the insertion limit curves, Figures 3.1-1, 3.1-2, 3.1-3, 3.1-4, 3.1-5 and 3.1-6, are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. Operation outside of the power-imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power-imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the insertion limits, as defined by Figures 3.1-1, 3.1-2, 3.1-3, 3.1-4, 3.1-5, and 3.1-6, and if a 4 percent QUADRANT POWER TILT exists. Additional conservatism is introducted by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

The conservative application of the above peaking augmentation factors compensates for the potential peaking penalty due to fuel rod bow.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensures that the original criteria are met.

The definitions of the design limit nuclear power peaking factors as used in these specifications are as follows:

F_Q Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

$F_{\Delta H}^{N}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met, provided:

$$F_Q \leq 3.12; \quad F_{\Delta H}^{N} \leq 1.78$$

Power Peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 6.5% (indicated position) from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained between + 6 percent and - 25 percent at RATED THERMAL POWER.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F_{\Delta H}^{N}$:

a. The measurement of total peaking factor, F_Q^{Meas}, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding 19.7 kw/ft locally, and from going below a minimum DNBR of 1.30 by automatic protection on power, AXIAL POWER IMBALANCE, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the Reactor Protection System.

The QUADRANT POWER TILT limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The QUADRANT POWER TILT limit at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. The limit was selected to provide an allowance for the uncertainty associated with the power tilt. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of the limit.

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 FSAR initial assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.3 INSTRUMENTATION

BASES

3/4.3.1 and 3/4.3.2 REACTOR PROTECTION SYSTEM (RPS) AND ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) INSTRUMENTATION

The OPERABILITY of the RPS and ESFAS instrumentation systems ensure that 1) the associated ESFAS action and/or RPS trip will be initiated when the parameter monitored by each channel or combination thereof 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 RPS and ESFAS 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 RPS and ESFAS action function associated with each channel is completed within the time limit assumed in the safety analyses. No credit was taken in the analyses for those channels with response times indicated as not applicable.

Response time may be demonstrated by any series of sequential, overlapping or total channel test measurements provided that such test demonstrate the total channel response time as defined. Sensor response time verification may be demonstrated by either 1) in place, onsite or offsite test measurements or 2) utilizing replacement sensors with certified response times.

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

3/4.3.3.2 INCORE DETECTORS

The OPERABILITY of the incore detectors ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. See Bases Figures 3-1 and 3-2 for examples of acceptable minimum incore detector arrangements.

3/4.3.3.3 SEISMIC INSTRUMENTATION

The OPERABILITY of the seismic instrumentation ensures that sufficient capability is available to promptly determine the magnitude of a seismic event so that the response of those features important to safety may be evaluated. This capability is required to permit comparison of the measured response to that used in the design basis for the facility. This instrumentation is consistent with the recommendations of Safety Guide 12 "Instrumentation for Earthquakes", March 1971.

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. This instrumentation is consistent with the recommendations of Regulatory Guide 1.23 "Onsite Meteorological Programs", February 1972.

3/4.3.3.5 REMOTE SHUTDOWN INSTRUMENTATION

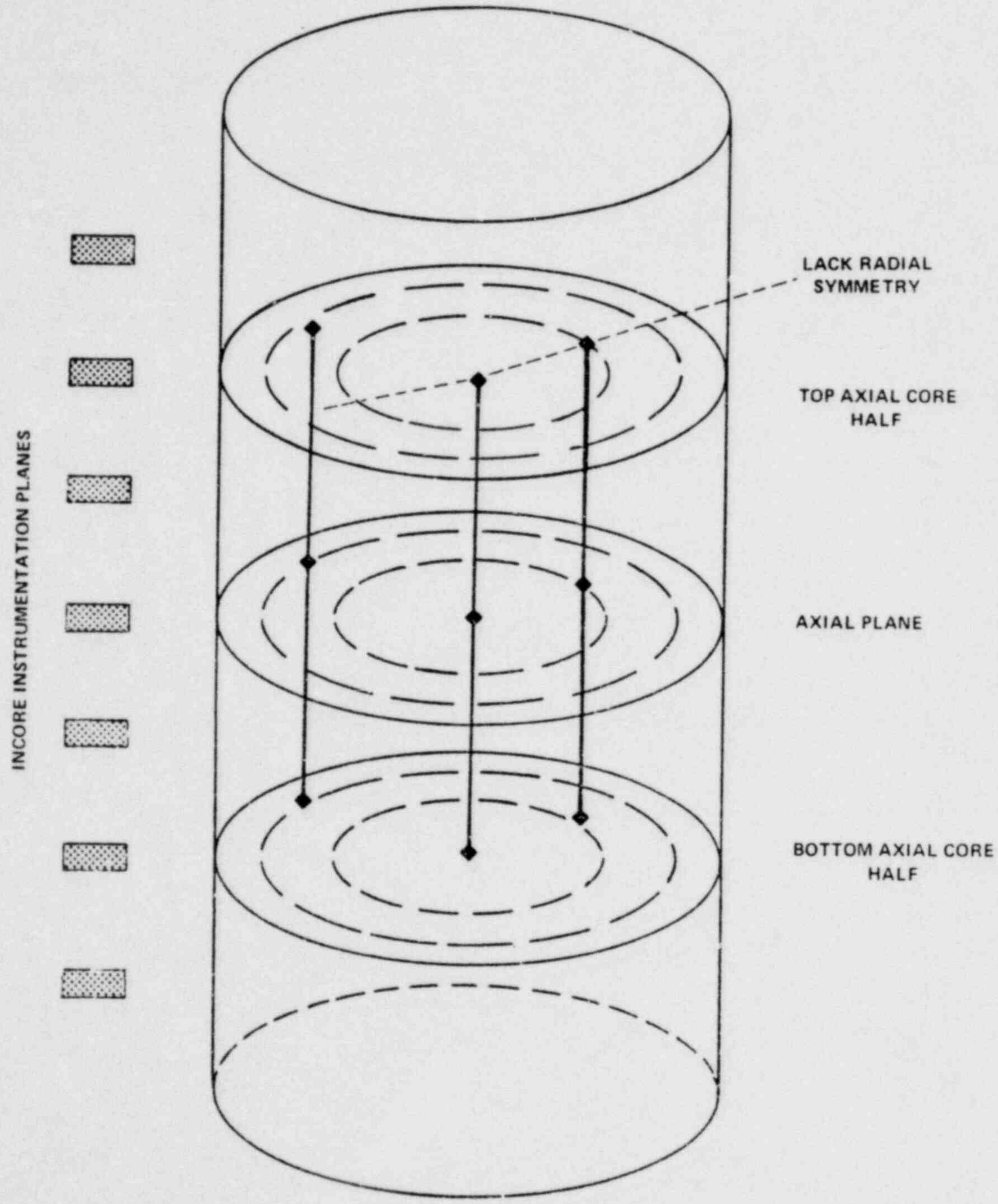
The OPERABILITY of the remote shutdown instrumentation ensures that sufficient capability is available to permit shutdown and maintenance of

REMOTE SHUTDOWN INSTRUMENTATION (Continued)

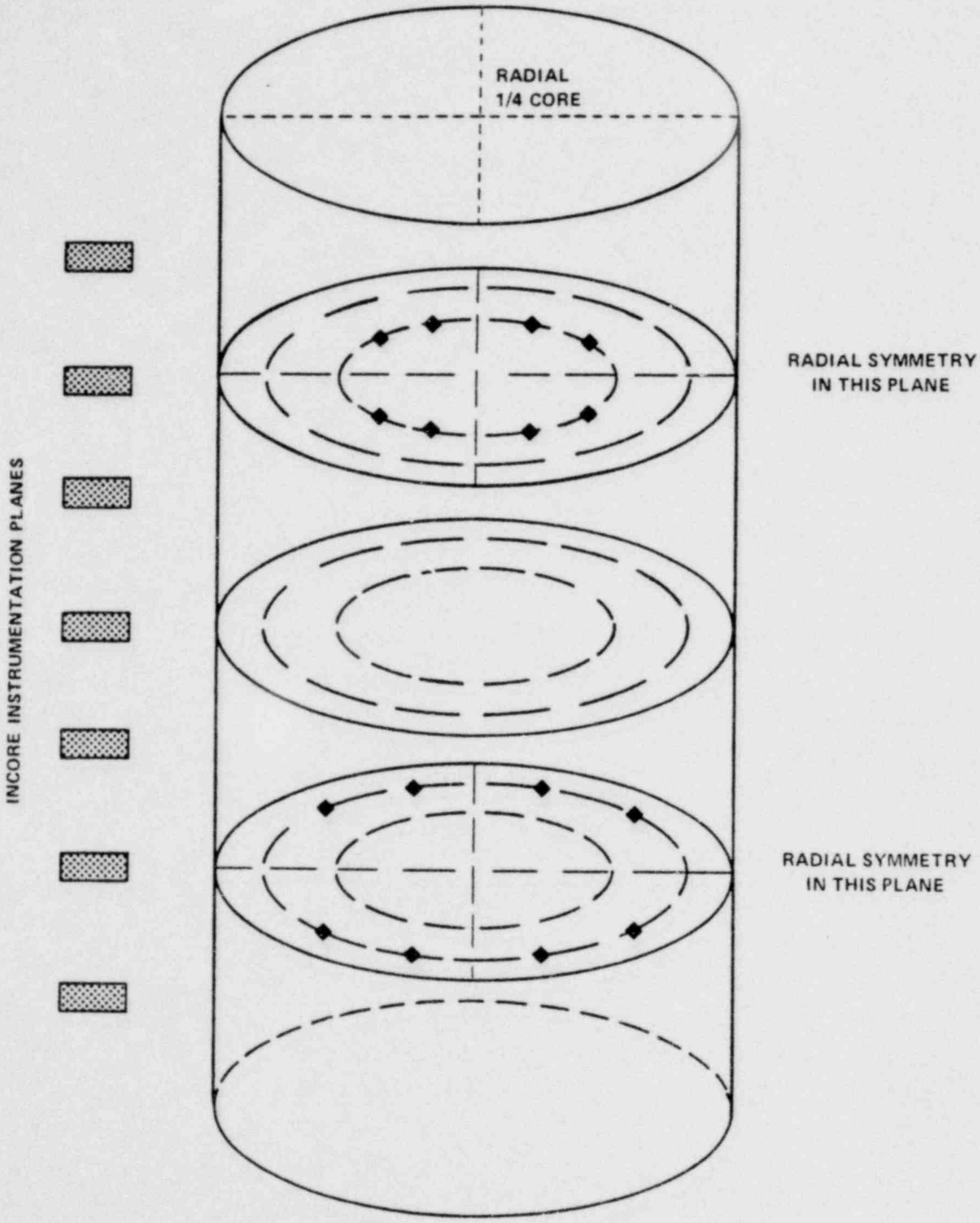
HOT STANDBY of the facility from locations outside of the control room. This capability is required in the event control room habitability is lost and is consistent with General Design Criterion 19 of Appendix "A", 10 CFR 50.

3/4.3.3.6 POST-ACCIDENT INSTRUMENTATION

The OPERABILITY of the post-accident instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident. This capability is consistent with the recommendations of Regulatory Guide 1.97, "Instrumentation for Light-Water Cooled Nuclear Power Plants to assess Plant Conditions During and Following an Accident", December 1975.



Bases Figure 3-1 Incore Instrumentation Specification
Acceptable Minimum AXIAL POWER IMBALANCE Arrangement



Bases Figure 3-2 Incore Instrumentation Specification
Acceptable Minimum QUADRANT POWER TILT Arrangement

3/4.4.1 REACTOR COOLANT LOOPS

The plant is designed to operate with both reactor coolant loops in operation, and maintain DNBR above 1.30 during all normal operations and anticipated transients. With one reactor coolant pump not in operation in one or both loops, THERMAL POWER is restricted by the Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE, ensuring that the DNBR will be maintained above 1.30 at the maximum possible THERMAL POWER for the number of reactor coolant pumps in operation or the local quality at the point of minimum DNBR equal to 15%, whichever is more restrictive.

A single reactor coolant loop provides sufficient heat removal capability for removing core decay heat while in HOT STANDBY; however, single failure considerations require placing a DHR loop into operation in the shutdown cooling mode 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 psig. Each safety valve is designed to relieve 317,973 lbs per hour of saturated steam at the valve's 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 DHR 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 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from any transient.

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 Code.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valves against water relief.

The low level limit is based on providing enough water volume to prevent a pressurizer low level or a reactor coolant system low pressure condition that would actuate the Reactor Protection System or the Engineered Safety Feature Actuation System as a result of a reactor scram. The high level limit is based on maximum reactor coolant inventory assumed in the safety analysis.

The power operated relief valves and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valves minimizes the undesirable opening of the spring-loaded pressurizer code safety valves.

3/4.4.5 STEAM GENERATORS

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those chemistry limits found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these chemistry 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 GPM). 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 GPM can be detected by monitoring the secondary coolant. Leakage in excess of this limit will require plant shutdown and an unscheduled inspection, during which the leaking tubes will be located and plugged.

Wastage-type defects are unlikely with proper chemistry treatment of the secondary coolant. However, even if a defect should develop in service, it will be found during scheduled inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit of 40% of the tube nominal wall thickness. Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be promptly reported to the Commission pursuant to Specification 6.9.1 prior 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.

The steam generator water level limits are consistent with the initial conditions assumptions in the FSAR.

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 detect and monitor 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 OPERATIONAL LEAKAGE

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.

Industry experience has shown that, while a limited amount of leakage is expected from the RCS, the UNIDENTIFIED LEAKAGE portion of this 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 total steam generator tube leakage limit of 1 GPM for all steam generators ensures that the dosage contribution from tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 GPM limit is consistent with the assumptions used in the analysis of these accidents.

The 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 CONTROLLED LEAKAGE limit of 10 GPM restricts operation with a total RCS leakage from all RC pump seals in excess of 10 GPM.

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 shown on Table 3.4-1 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 the Part 100 limit 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. 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 the specific site parameters of the 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.

2 ∞

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 I-131 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.

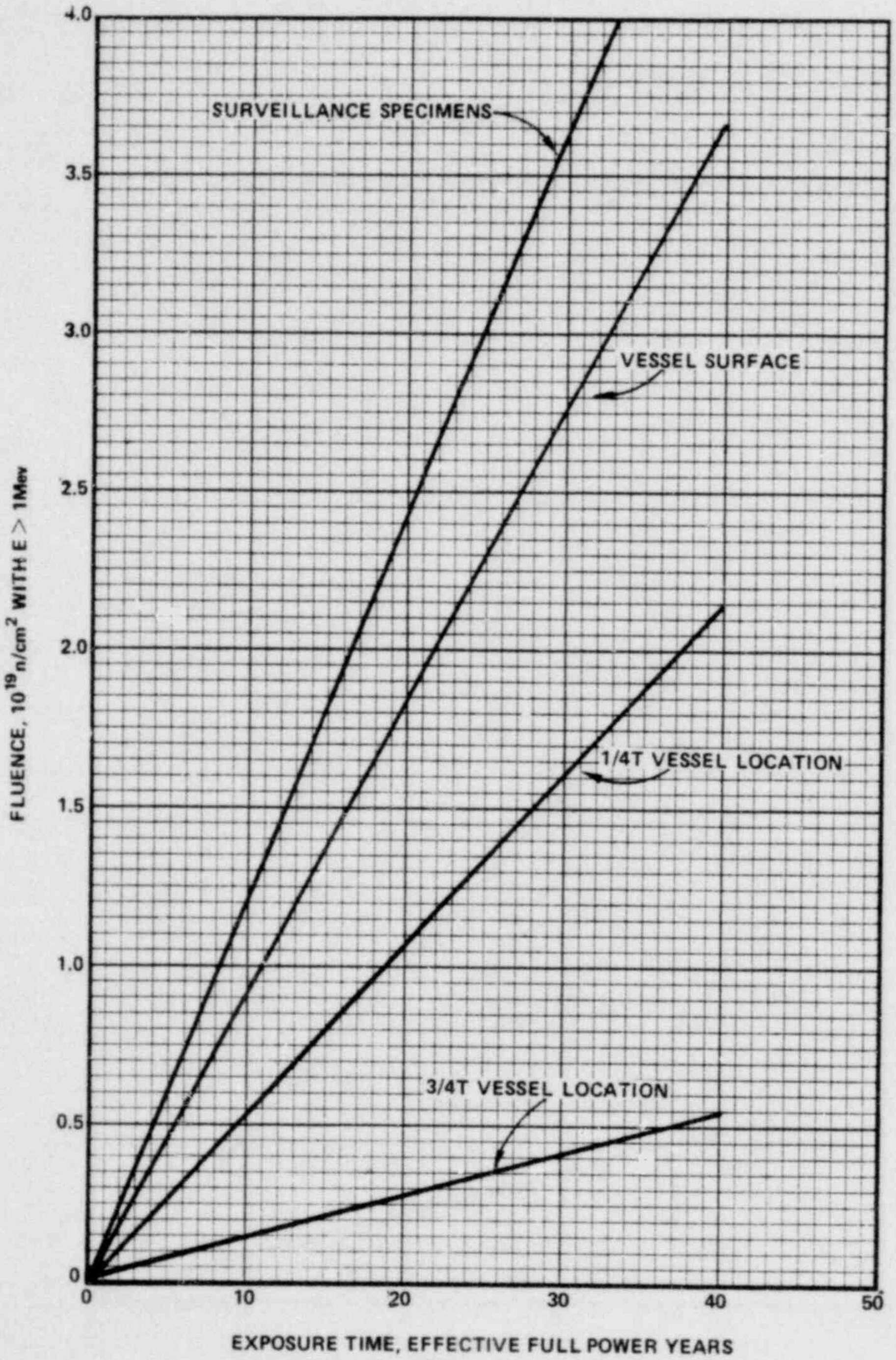
Reducing T_{avg} to < 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves.

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.

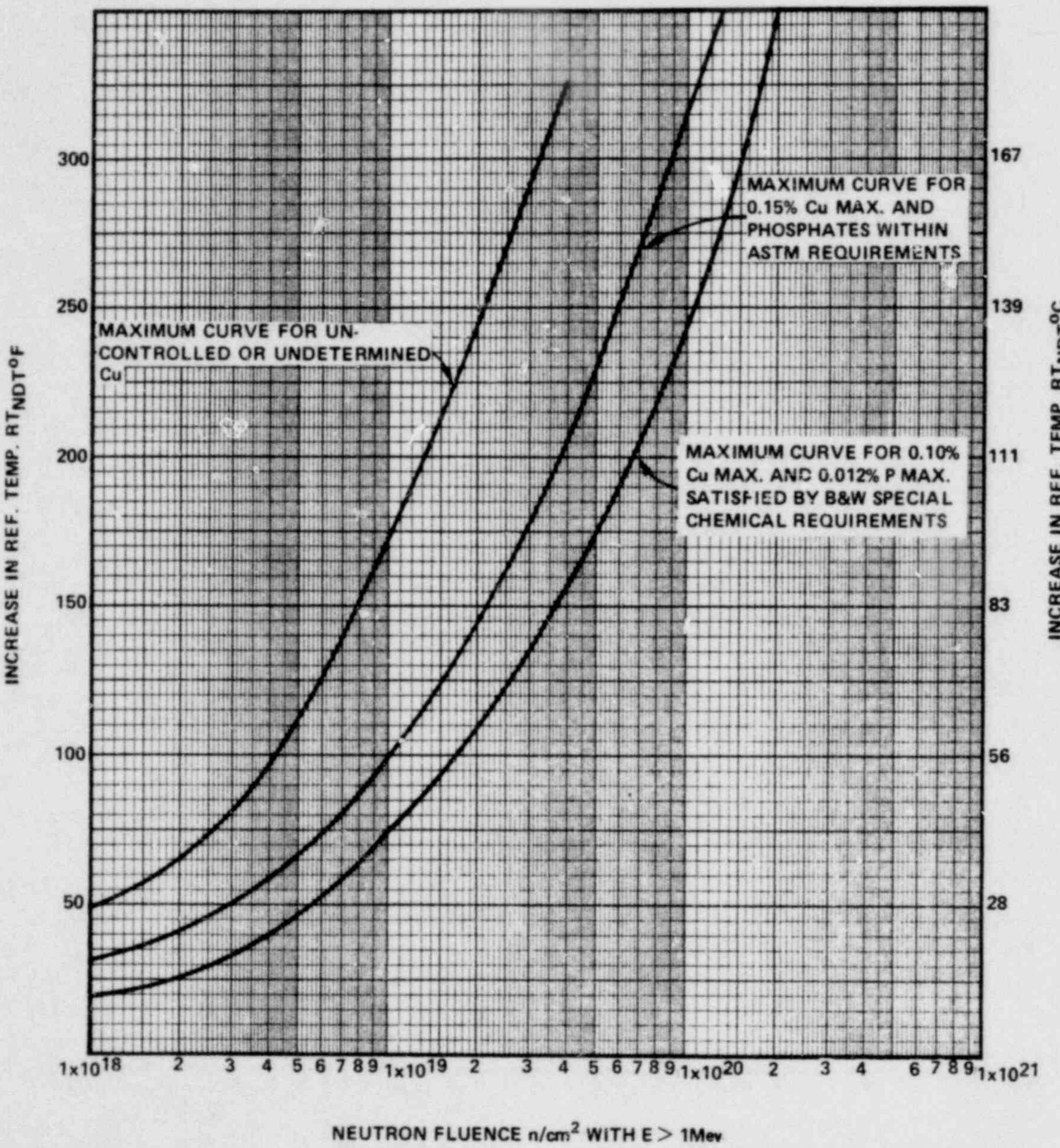
3/4.4.9 PRESSURE/TEMPERATURE LIMITS

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 4.1.2.4 of the FSAR. During heatup and cooldown, 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.



Bases Figure 4-1 Fast Neutron Fluence (E > 1 mev) as a Function of Full Power Service Life



Bases Figure 4-2 Effect of Fluence and Copper Content on Shift of RT_{NDT} for Reactor Vessel Steels Exposed to 550°F Temperature

BASES TABLE 4-1

REACTOR VESSEL TOUGHNESS

COMPONENT	MATERIAL TYPE	CU %	P %	S %	RT_{NDT} F	TRANS UPPER SHELF FT-LB	ADJUSTED RT_{NDT} FOR 5 FULL POWER YEARS @ 1/4 T, °F	ADJUSTED RT_{NDT} FOR 5 FULL POWER YEARS @ 3/4 %, °F
Nozzle Belt	SA-508 Cl.2	.054	.008	.006	+10	183	40	26
*Upper Shell	SA-533B	.20	.008	.016	+20	88	95	58
**Upper Shell	SA-533B	.20	.008	.016	+20	90	95	58
Lower Shell	SA-533B	.12	.013	.015	-20	119	25	01
Lower Shell	SA-533B	.12	.013	.015	+45	88	90	66
***Surveillance	Weld	.30	.020	.005	+43	63	NA	NA
Upper Long (40%)	Weld	.20	.009	.009	(+20)****	66****	90	54
Upper Long (40%)	Weld	.105	.091	.004	(+20)****	66****	48	34
Upper Circum (60%)	Weld	.106	.014	.013	(+20)****	66****	65	[illegible]
Upper Circum (40%)	Weld	.19	.021	.016	(+20)****	66****	95	58
Middle Circum (100%)	Weld	.27	.014	.011	(+20)****	66****	95	58
Lower Long (100%)	Weld	.22	.015	.013	(+20)****	66****	84	52
Lower Circum (100%)	Weld	.20	.015	.021	(+20)****	66****	20	20
Out 1st Nozzle	Weld	.19	.021	.016	(+20)****	66****	20	20

* Surveillance Base Metal A
** Surveillance Base Metal B
*** Surveillance Weld
**** Estimated Value

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Consequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The heatup limit curve, Figure 3.4-2 is a composite curve which was prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate up to 100°F per hour. The cooldown limit curve, Figure 3.4-3 is a composite curve which was prepared based upon the same type analysis with the exception that the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stresses at the outside wall. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of 5 EFPY.

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in BASES Table 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 and copper content of the material in question, can be predicted using BASES Figures 4-1 and 4-2. The heatup and cooldown limit curves, of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of 5 EFPY, 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 the radius are essentially identical, the measured transition shift for a sample can 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 closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). This region largely controls the pressure-temperature limitations of the first several service periods. The outlet nozzles of the reactor vessel also affect the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} temperature of the beltline region materials will be high enough so that the beltline region of the reactor vessel will start to control the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves are es[illegible], the maximum allowable pressure as a function of fluid tempe[illegible]e is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The calculated pressure temperature limit curves are then adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of the fifth effective full power year. The fifth effective full power year was selected because the second surveillance capsule will be withdrawn at the end of the fifth cycle. The time difference between the fifth cycle and fifth effective full power year provides adequate time for establishing the operating pressure and temperature limitations for the period of operation after the fifth effective full power year.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, 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 the radius are essentially identical, the measured transition shift for a sample can be applied

BASES

with confidence to the adjacent section of the reactor vessel. The limit 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. Since the first surveillance program capsule will be withdrawn at the end of the first effective full power year and the limit curves were prepared based upon the predicted impact properties at the end of the fifth effective full power year, it is predicted that no readjustment will be required to the limit curves for the first 5 effective full power years. Adjustment may be required after the withdrawal of the second capsule.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The unirradiated impact properties and residual elements of the beltline region materials are listed in Bases Table 4-1. The adjusted reference temperature are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated. The predicted ΔRT_{NDT} are calculated using the respective neutron fluence and copper and phosphorus contents. Bases Figure 4-1 illustrates the calculated peak neutron fluence, at several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules as a function of exposure time.

Bases Figure 4-2 illustrates the design curves for predicting the radiation induced ΔRT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. The adjusted RT_{NDT}'s of the beltline region materials at the end of the fifth full power year are listed in Bases Table 4-1. The adjusted RT_{NDT}'s are given for the 1/4T and 3/4T (T is wall thickness) vessel wall locations. The assumed RT_{NDT} of the closure head region and of the outlet nozzle steel forgings is 60°F.

During cooldown at the higher temperatures, the limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are the vertical segments of the limit lines on Figures 3.4-2 and 3.4-4, respectively. These limits will not require adjustments due to the neutron fluences.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

BASES

All pressure-temperature limit curves are applicable up to the fifth effective full power year. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, and 3.4-3, and 3.4-4.

The pressure and temperature limits shown on Figures 3.4-2 and 3.4-3 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 number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown 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 inspection programs for ASME Code Class 1, 2 and 3 components, except steam generator tubes, 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 these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrates that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

BASES

3/4.5.1 CORE FLOODING TANKS

The OPERABILITY of each core flooding tank ensures that a sufficient volume of borated water will be immediately forced into the reactor vessel in the event the RCS pressure falls below the pressure of the tanks. This initial surge of water into the vessel provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on volume, boron concentration and pressure ensure that the assumptions used for core flooding tank injection in the safety analysis are met.

The limits for operation with a core flooding tank inoperable for any reason except an isolation valve closed minimizes the time exposure of the plant to a LOCA event occurring concurrent with failure of an additional tank which may result in unacceptable peak cladding temperatures. If a closed isolation valve cannot be immediately opened, the full capability of one tank is not available and prompt action is required to place the reactor in a mode where this capability is not required.

BASES

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two independent ECCS subsystems with RCS average temperature > 280°F 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 core flooding tanks is capable of supplying sufficient core cooling to maintain 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.

With the RCS temperature below 280°F, one OPERABLE ECCS subsystem is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the limited core cooling requirements.

The Surveillance Requirements provided to ensure OPERABILITY of each component ensures, that, at a minimum, the assumptions used in the safety analyses are met and that subsystem OPERABILITY is maintained. The decay heat removal system leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase of the low pressure injection will not be exceeded.

3/4.5.4 BORATED WATER STORAGE TANK

The OPERABILITY of the borated water storage tank (BWST) 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 BWST 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 BWST 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.

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The limits on contained water volume, and boron concentration ensure a pH value of between 7.2 and 11.0 of the solution sprayed within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 49.6 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Appendix "J" of 10 CFR 50.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on CONTAINMENT INTEGRITY and containment leak rate given. Surveillance testing of the air lock seals provide assurance that the overall air lock leakage will not become excessive due to seal damage during the intervals between air lock leakage tests.

3/4.6.1.4 INTERNAL PRESSURE

The limitations on containment internal pressure ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 3.0 psig and 2) the containment peak pressure does not exceed the design pressure of 54.6 psig during LOCA conditions.

The maximum peak pressure obtained from a LOCA event is 49.6 psig. The limit of 3 psig for initial positive containment pressure will limit the total pressure to 52.6 psig which is less than the design pressure and is consistent with the safety analyses.

3/4.6.1.5 AIR TEMPERATURE

The limitations on containment average air temperature ensure that the overall containment average air temperature does not exceed the initial temperature condition assumed in the accident analysis for a LOCA.

3/4.6.1.6 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 52.6 psig in the event of a LOCA. The measurement of containment tendon lift off force, the visual and metallurgical examination of tendons, anchorages and liner, and the Type A leakage tests are 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 capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses. The leak rate surveillance requirements assure that the leakage rates assumed for the system during the recirculation phase will not be exceeded.

3/4.6.2.2 SPRAY ADDITIVE SYSTEM

The OPERABILITY of the spray additive system ensures that sufficient NaOH and $Na_2S_2O_3$ are added to the containment spray in the event of a LOCA. The minimum $Na_2S_2O_3$ volume and concentration ensures sufficient $Na_2S_2O_3$ is available to remove organic iodine from the containment atmosphere and return it to the spray water. The limits on contained sodium hydroxide solution volume and concentration, and contained sodium thiosulfate solution volume and concentration ensure a pH value of between 7.2 and 11.0 of the solution sprayed within containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. These assumptions are consistent with the iodine removal efficiency assumed in the safety analyses.

3/4.6.2.3 CONTAINMENT COOLING SYSTEM

The OPERABILITY of the containment cooling system ensures that 1) the containment air temperature will be maintained within limits during normal operation, and 2) adequate heat removal capacity is available when operated in conjunction with the containment spray systems during post-LOCA conditions.

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. The purge system is capable of controlling the expected hydrogen generation associated with 1) zirconium-water reactions, 2) radiolytic decomposition of water and 3) corrosion of metals within containment. These hydrogen control systems are consistent with the recommendations of Regulatory Guide 1.7, "Control of Combustible Gas Concentrations in Containment Following a LOCA", March 1971.

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 1050 psig during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition. The total relieving capacity for all valves on all of the steam lines is 13,007,774 lbs/hr which is 122.7 percent of the total secondary steam flow of 10.6×10^6 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 the allowable THERMAL POWER restriction in Table 3.7-1.

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 Nuclear Overpower channels. The reactor trip setpoint reductions are derived on the following bases:

$$SP = \frac{(X) - (Y)(V)}{X} \times 105.5$$

where:

SP = reduced Nuclear Overpower Trip Setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam generator

105.5 = Nuclear Overpower Trip Setpoint specified in Table 2.2.1

X = Total relieving capacity of all safety valves per steam generator in lbs/hour

Y = Maximum relieving capacity of any one safety valve in lbs/hour

3/4.7.1.2 EMERGENCY FEEDWATER SYSTEMS

The OPERABILITY of the emergency feedwater systems ensures that the Reactor Coolant System can be cooled down to less than 280°F from normal operating conditions in the event of a total loss of offsite power.

The electric driven emergency feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1144 psig to the entrance of the steam generators. Each steam driven emergency feedwater pump is capable of delivering a total feedwater flow of 740 gpm at a pressure of 1144 psig to the entrance of the steam generators. This capacity is sufficient to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 280°F where the Decay Heat Removal System may be placed into operation.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 280°F in the event of a total loss of offsite power or of the main feedwater system. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 24 hours with steam discharge to atmosphere concurrent with loss of offsite power. The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant offsite radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. This dose includes the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line. These values are consistent with the assumptions used in the safety analyses.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will 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 safety analyses.

3/4.7.1.6 SECONDARY WATER CHEMISTRY

A test program will be conducted during approximately the first 6 months following 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 parameters and the frequencies for monitoring these parameters.

The test program will include an analysis of the chemical constitutents of the condenser cooling water at the point of intake. The analysis shall identify the various traces of ions which upon concentration in the condensate 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 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 110°F and 237 psig are based on a steam generator RT_{NDT} of 40°F and are sufficient to prevent brittle fracture.

3/4.7.3 CLOSED CYCLE COOLING WATER SYSTEMS

The OPERABILITY of the closed cycle cooling water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.4 SEA WATER SYSTEM

The OPERABILITY of the sea water system ensures that sufficient cooling capacity is available for continued operation of safety related equipment during normal and accident conditions. The redundant cooling capacity of this system, assuming a single failure, is consistent with the assumptions used in the safety analyses.

3/4.7.5 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 a 30 day cooling water supply to safety related equipment without exceeding their design basis temperature and is consistent with the recommendations of Regulatory Guide 1.27, "Ultimate Heat Sink for Nuclear Plants", March 1974.

3/4.7.6 FLOOD PROTECTION

The ACTION tc be taken in the event of flood together with a Hurricane Warning ensures that the facility will be placed in a safe condition. The limit of evaluation 98 feet Plant Datum is based on the maximum elevation at which facility flood control measures provide protection to safety related equipment.

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 for operations personnel during and following all credible accident conditions. The control room emergency ventilation system is not required to maintain a positive pressure. The safety analysis of the system assumed slight in-leakage to the control room which would be subsequently removed by the full recirculation flow. The OPERABILITY of this system in conjunction with control room design provisions is based on limiting the radiation exposure to personnel occupying the control room to 5 rem or less whole body, or its equivalent. This limitation is consistent with the requirements of General Design Criterion 19 of Appendix "A", 10 CFR 50

3/4.7.8 AUXILIARY BUILDING VENTILATION EXHAUST SYSTEM

The OPERABILITY of the auxiliary building ventilation exhaust 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 were assumed in the safety analyses.

3/4.7.9 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.

The inspection frequency applicable to snubbers containing seals fabricated from materials which have been demonstrated compatible with their operating environment 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 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.

3/4.7.10 SEALED SOURCE CONTAMINATION

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values.

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 Criterion 17 of Appendix "A" to 10 CFR 50.

The ACTION requirements specified for the levels of degradation of the power sources provide restriction upon continued facility operation commensurate with the level of degradation. The OPERABILITY of the power sources are consistent with the initial condition assumptions of the safety analyses and are based upon maintaining at least one 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.

3/4.9 REFUELING OPERATIONS

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on minimum boron concentration 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 of a K_{eff} of no greater than 0.95, which includes a conservative allowance for uncertainties, is sufficient to prevent reactor criticality during refueling operations.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the safety 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 requirements 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 FUEL HANDLING BRIDGE OPERABILITY

The OPERABILITY requirements of the hoist bridges used for movement of fuel assemblies ensures that: 1) fuel handling bridges will be used for movement of control rods and fuel assemblies, 2) each hoist has sufficient load capacity to lift a fuel element, and 3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE BUILDING

The restriction on movement of loads in excess of the nominal weight of a fuel and control rod assembly and associated handling tool over other fuel assemblies in the storage pool ensures that in the event this load is dropped (1) the activity release will be limited to that contained in a single fuel assembly, and (2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the accident analyses.

3/4.9.8 COOLANT CIRCULATION

The requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during the REFUELING MODE, and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effect of a boron dilution incident and prevent boron stratification.

3/4.9.9 CONTAINMENT PURGE AND EXHAUST ISOLATION SYSTEM

The OPERABILITY of this system ensures that the containment purge and exhaust penetrations 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 WATER LEVEL - REACTOR VESSEL 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 safety analysis.

BASES

3/4.9.11 STORAGE POOL

The requirement for missile shields to be installed over the storage pool insures that the tornado missile protection assumptions are satisfied.

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the safety analysis.

3/4.9.12 STORAGE POOL VENTILATION

The requirement for the auxiliary building ventilation exhaust system servicing the storage pool area to be OPERABLE 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 safety analyses.

3/4.10 SPECIAL TEST EXCEPTIONS

BASES

3/4.10.1 GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

This special test exception permits individual control rods to be positioned outside of their specified group heights and insertion limits and to be assigned to other than specified control rod groups, and permits AXIAL POWER IMBALANCE and QUADRANT POWER TILT limits to be exceeded during the performance of such PHYSICS TESTS as those required to 1) measure control rod worth, 2) determine the reactor stability index and damping factor under xenon oscillation conditions and 3) calibrate AXIAL POWER IMBALANCE and QUADRANT POWER TILT instrumentation.

3/4.10.2 PHYSICS TESTS

This special test exception permits PHYSICS 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.

3/4.10.3 NO FLOW TESTS

This special test exception permits reactor criticality under no flow conditions and is required in order to perform certain startup and PHYSICS TESTS while at low THERMAL POWER levels.

3/4.10.4 SHUTDOWN MARGIN

This special test exception provides that a minimum amount of control rod worth is immediately available for reactivity control when tests are performed for control rod 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.

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.

5.2 CONTAINMENT

CONFIGURATION

5.2.1 The reactor containment building is a steel lined, reinforced concrete building of cylindrical shape, with a dome roof and having the following design features:

a. Nominal inside diameter = 130 feet.

b. Nominal inside height = 157 feet.

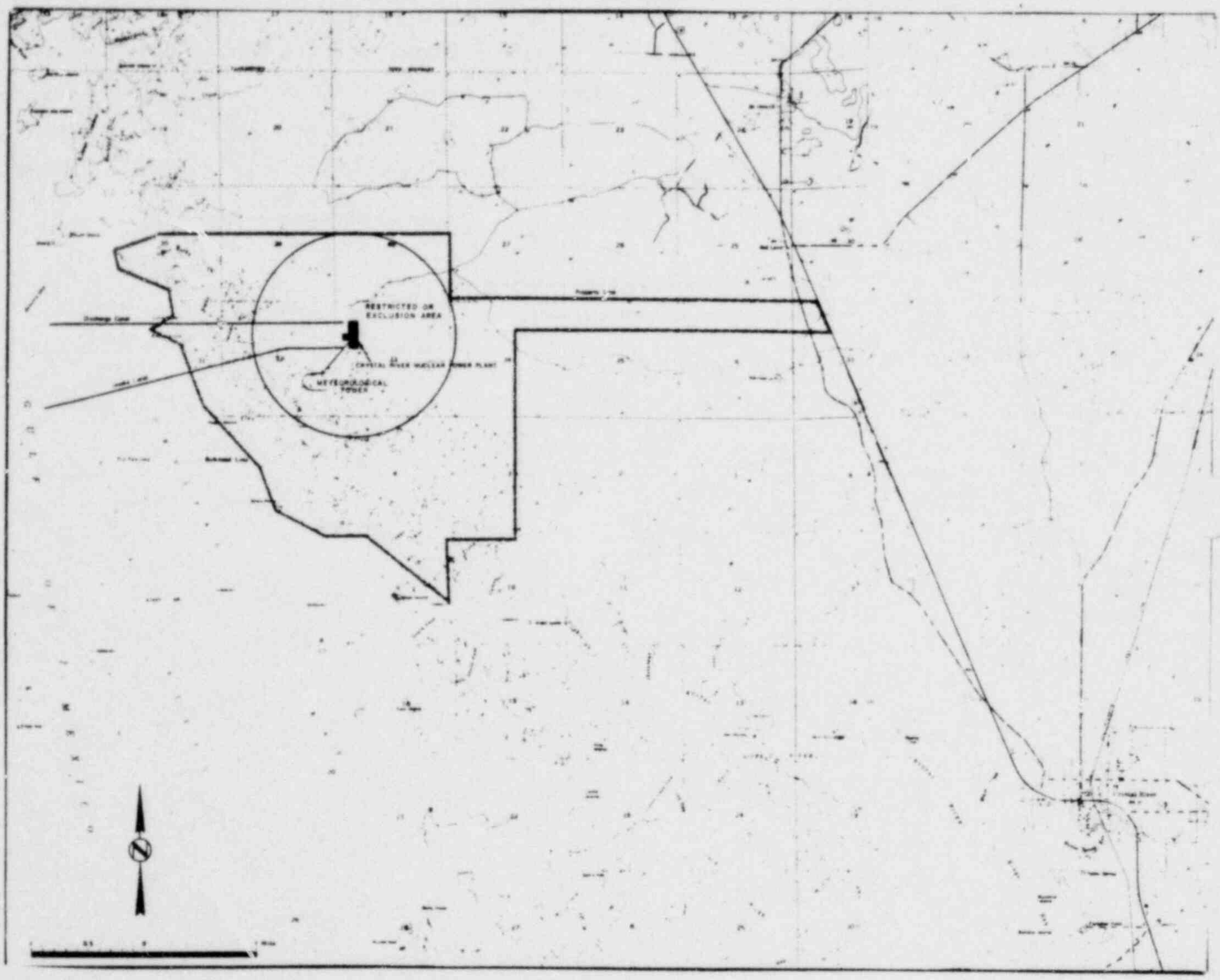
c. Minimum thickness of concrete walls = 3.5 feet.

d. Minimum thickness of concrete roof = 3 feet.

e. Minimum thickness of concrete floor pad = 12.5 feet.

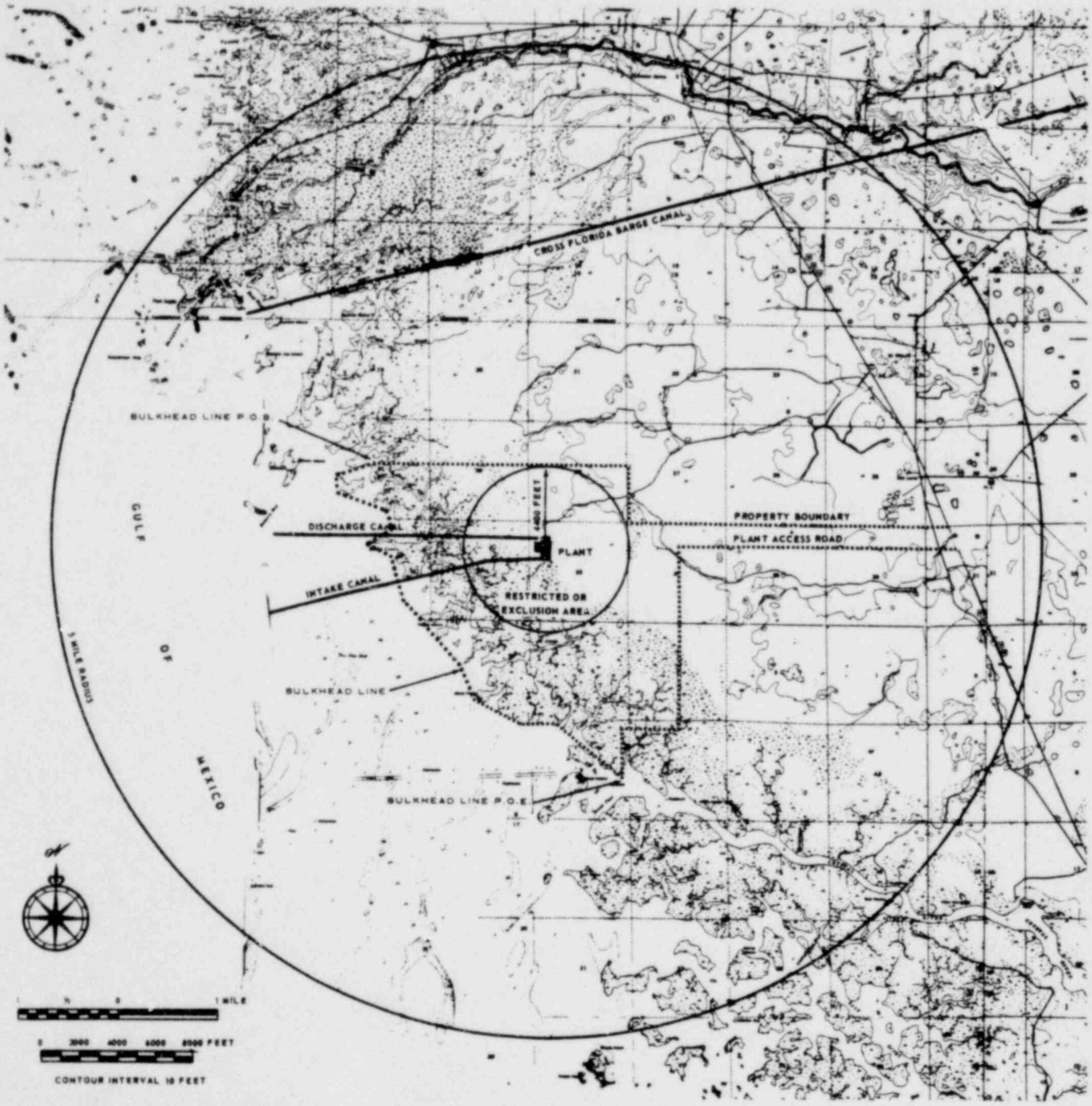
f. Nominal thickness of steel liner = 3/8 inches.

g. Net free volume = 2×10^6 cubic feet.



<u>EXCLUSION AREA</u>

<u>FIGURE 5.1-1</u>



LOW POPULATION ZONE

FIGURE 5.1-2

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 55 psig and a temperature of 281°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2229 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 a maximum enrichment of 3.50 weight percent U-235.

The first cycle fuel loading shall contain 68 burnable poison rod assemblies with each assembly containing up to 16 burnable poison rods of sintered Al_2O_3-B_4C clad with Zircaloy-4.

CONTROL RODS

5.3.2 The reactor core shall contain 61 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The APSR's shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

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 4.1.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,180 $\pm$ 200 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new and spent fuel storage racks are designed and shall be maintained with a nominal 21 1/8 inch center-to-center distance between fuel assemblies placed in the 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 a conservative allowance of >1% $\Delta k/k$ for uncertainties as described in Section 3.2 of the FSAR.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 138 feet 4 inches.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 240 fuel assemblies and 15 failed fuel containers.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limit of Table 5.7-1.

TABLE 5.7-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

Component or System	Cycle or Transient Limit	Design Cycle or Transient
1. Reactor Coolant System	240 Heatup and Cooldown Cycles	70°F to 557°F to 70°F
2. Reactor Coolant System	160 Step Load Reduction Cycles (Resulting from turbine trip)	100% to 8% RTP*
3. Reactor Coolant System	150 Step Load Reduction Cycles (Resulting from electrical load rejection)	100% to 8% RTP*
4. Reactor Coolant System	40 Reactor Trip Cycles (Resulting from loss of electric power to all RC pumps)	Reactor Trip
5. Reactor Coolant System	160 Reactor Trip Cycles (Resulting from turbine trip without automatic control action)	Reactor Trip
6. Reactor Coolant System	40 Reactor Trip Cycles (Resulting from rod withdrawal accident)	Reactor Trip
7. Once Through Steam Generator	88 Reactor Trip Cycles (Resulting from complete loss of all main feed-water)	Reactor Trip
8. Once Through Steam Generator	40 Reactor Trip Cycles (Resulting from loss of station power)	Reactor Trip
9. Once Through Steam Generator	20 Reactor Trip Cycles (Resulting from loss of feedwater to one steam generator)	Reactor Trip

*RATED THERMAL POWER

TABLE 5.7-1 (Continued)

Component or System	Cycle or Transient Limit	Design Cycle or Transient
10. Once Through Steam Generator	10 Reactor Trip Cycles (Resulting from stuck open turbine bypass valve)	Reactor Trip
11. Reactor Coolant System	80 Rapid Depressurization	2200 psig to 300 psig in one hour
12. Reactor Coolant System	20 Change of Flow Cycles	Loss of one or more RC pumps
13. Reactor Coolant System	20 Hydrostatic Test	Pressurized to $\geq$ 3125 psig
14. Once Through Steam Generator	35 Hydrostatic Tests	Pressurized to $\geq$ 3125 psig
15. Reactor Coolant System	480 Test Transients	High Pressure Injection Test
16. Reactor Coolant System	240 Test Transients	Core Flooding Check Valve Test

SECTION 6.0

ADMINISTRATIVE CONTROLS

6.0 ADMINISTRATIVE CONTROLS

6.1 RESPONSIBILITY

6.1.1 The Nuclear Plant Superintendent 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.

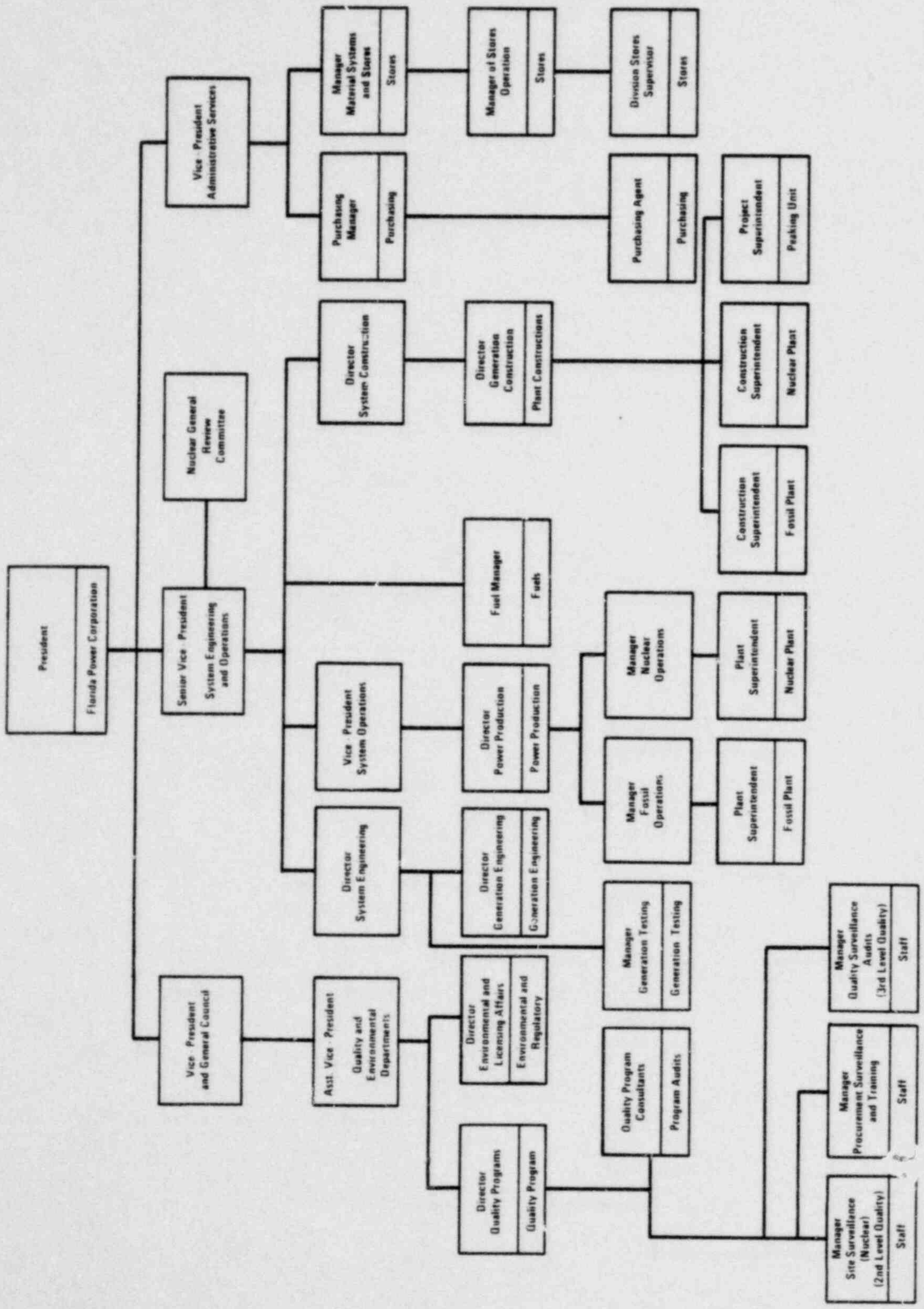
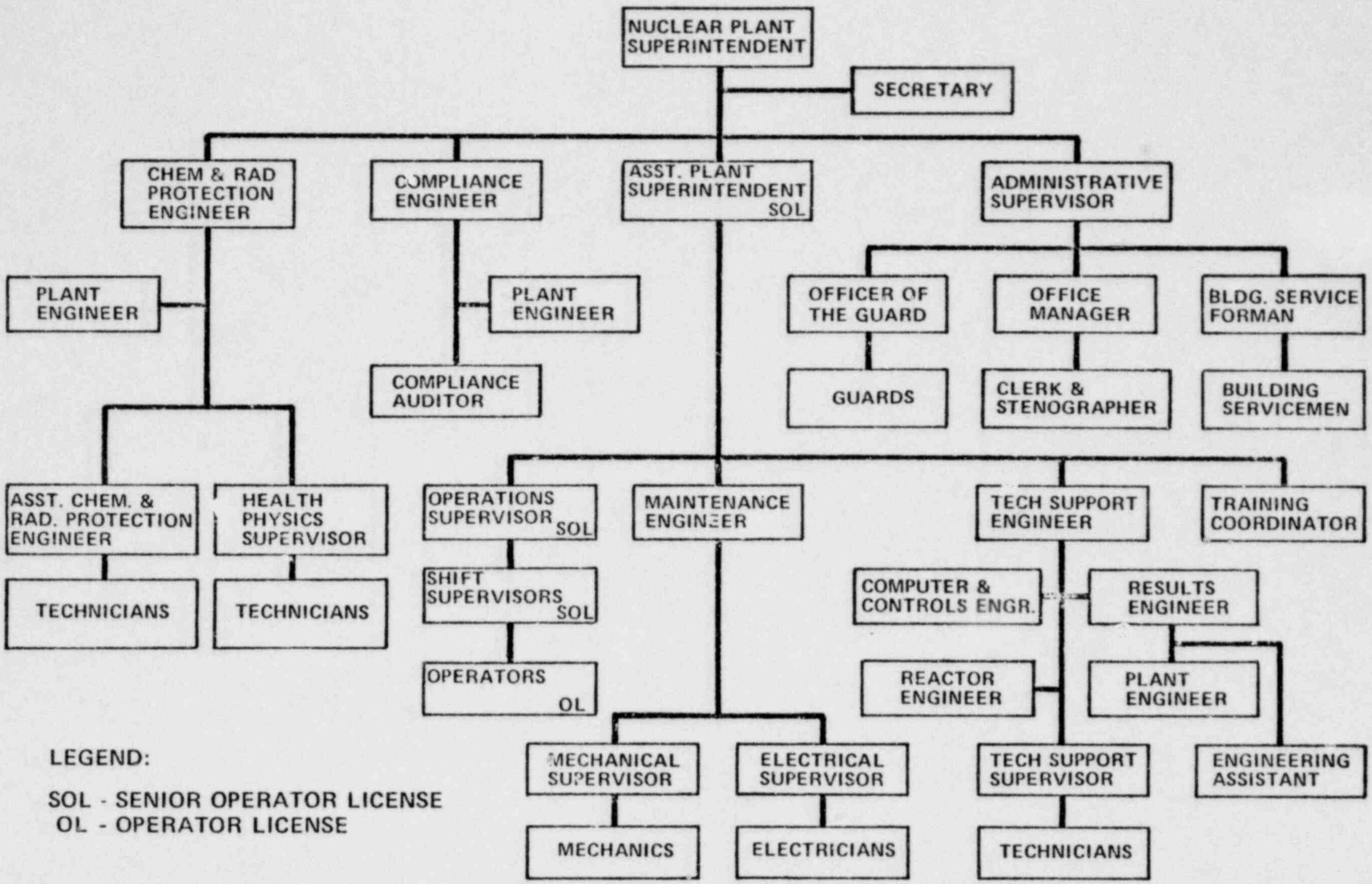


Figure 6.2-1
Offsite Organization



LEGEND:

SOL - SENIOR OPERATOR LICENSE
OL - OPERATOR LICENSE

FACILITY ORGANIZATION

FIGURE 6.2-2

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	3	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling Individual supervising CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less that the minumum requirement 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 Engineer 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 Assistant Nuclear Plant Superintendent 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 PLANT REVIEW COMMITTEE (PRC)

FUNCTION

6.5.1.1 The Plant Review Committee shall function to advise the Nuclear Plant Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Review Committee shall be composed of the:

Chairman:	Assistant Nuclear Plant Superintendent
Member:	Operations Supervisor
Member:	Technical Support Engineer
Member:	Maintenance Engineer
Member:	Chemistry and Radiation Protection Engineer

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in PRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRC shall meet at least once per calendar month and as convened by the PRC Chariman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PRC shall consist of the Chariman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Committee shall be responsbile 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 Nuclear Plant Superintendent to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear Safety.

c. Review of all proposed changes to the 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-Nuclear Operations and to the Chairman of the Nuclear General Review Committee.

f. Review of events requiring 24 hour written notification to the Commission.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Nuclear General Review Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear General Review Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear General Review Committee.

AUTHORITY

6.5.1.7 The Plant Review Committee shall:

a. Recommend to the Nuclear Plant Superintendent written approval or disapproval of items considered under 6.5.1(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 Manager-Nuclear Operations and the Nuclear General Review Committee of disagreement between the PRC and the Nuclear Plant Superintendent; however, the Nuclear Plant Superintendent shall have responsibility for resolution of such disagreements pursuant to 6.1.1 above.

RECORDS

6.5.1.8 The Plant Review Committee shall maintain written minutes of each meeting and copies shall be provided to the Manager-Nuclear Operations and Chairman of the Nuclear General Review Committee.

6.5.2 NUCLEAR GENERAL REVIEW COMMITTEE (NGRC)

FUNCTION

6.5.2.1 The Nuclear General Review Committee shall function to provide independent review and audit of designated activities in the areas of:

a. Nuclear power plant operations

b. Nuclear engineering

c. Chemistry and radiochemistry

d. Metallurgy

e. Instrumentation and control

f. Radiological safety

g. Mechanical and electrical engineering

h. Quality assurance practices

COMPOSITION

6.5.2.2 The NGRC shall be composed of the Chairman, Vice Chairman, and at least 5 members. No more than a minority of the members shall have line responsibility for operation of the facility. The committee shall collectively have the experience and competence required to review problems in the following areas:

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. Administrative controls

i. Environmental

j. Quality assurance practices

QUALIFICATIONS

6.5.2.3 The following minimum experience requirements shall be established for those persons involved in the independent off-site safety review and audit program:

a. Chairman and Vice-Chairman-Bachelor of Science in engineering or related field and ten years related experience including five years involvement with operation and/or design of nuclear power plants.

b. Member-Bachelor of Science in engineering or related field and five years related experience including three years involvement with operation and/or design of nuclear power plants.

ALTERNATES

6.5.2.4 All alternate members shall be appointed in writing by the NGRC Chairman to serve on a temporary basis; however no more than two alternates shall participate as voting members in NGRC activities at any one time.

CONSULTANTS

6.5.2.5 Consultants shall be utilized as determined by the NGRC Chairman to provide expert advice to the NGRC.

MEETING FREQUENCY

6.5.2.6 The NGRC 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

6.5.2.7 A quorum of NGRC shall consist of the Chairman or his designated alternate and five additional NGRC members, including alternates. No more than a minority of the quorum shall have line responsibility for operation of the facility.

REVIEW

6.5.2.8 The NGRC 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 this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or of internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

REVIEW (Continued)

g. Events requiring 24 hour written notification to the Commission.

h. All recognized indications of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the Plant Review Committee.

AUDITS

6.5.2.9 Audits of facility activities shall be performed under the cognizance of the NGRC. These audits shall encompass:

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable licence 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 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 activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B", 10 CFR 50, at least once per 24 months.

e. The 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 NRGC or the Senior Vice President-System Engineering and Operations.

AUTHORITY

6.5.2.10 The NGRC shall report to and advise the Senior Vice President-System Engineering and Operations on those areas of responsibility specified in Sections 6.5.2.8 and 6.5.2.9.

RECORDS

6.5.2.11 Records of NGRC activities shall be prepared, approved and distributed as indicated below:

a. Minutes of each NRGC meeting shall be prepared, approved and forwarded to the Senior Vice President-System Engineering and Operations within 14 days following each meeting.

b. Reports of reviews encompassed by Section 6.5.2.8 above, shall be prepared, approved and forwarded to the Senior Vice President-System Engineering and Operations within 14 days following completion of the review.

c. Audit reports encompassed by Section 6.5.2.9 above, shall be forwarded to the Senior Vice President-System Engineering and Operations 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 PRC and submitted to the NGRC and the Manager-Nuclear Operations.

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 Manager-Nuclear Operations and to the NGRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PRC. 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 NRGC and the Manager-Nuclear Operations within 14 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 PRC and approved by the Nuclear Plant Superintendent prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PRC and approved by the Nuclear Plant Superintendent within 14 days of implementation.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS AND REPORTABLE OCCURRENCE REPORTS

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, Region II, 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. ECCS Actuation, Specifications 3.5.2 and 3.5.3.

b. Inoperable Seismic Monitoring Instrumentation, Specification 3.3.3.3.

c. Inoperable Meteorological Monitoring Instrumentation, Specification 3.3.3.4.

d. Seismic event analysis, Specification 4.3.3.3.2.

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 a each power level.

b. and logs of principal maintenance activities, inspections, nd replacement of principal items of equipment related afety.

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 and fission detector 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 5.7-1.

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 PRC and the NGRC.

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 6.13.1a 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[7]	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 (* or schedule superseding for equipment type listed)
I. AIR-PURIFYING RESPIRATORS			
Facepiece, half-mask[4]	NP	5	30 CFR Part 11 Subpart K
Facepiece, full	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	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	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 (Continued)

(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 currently available for this equipment. Equipment must be evaluated by testing or on basis of available test information.

(7) Only for shaven faces and where nothing interferes with the seal of tight-fitting facepieces against the skin. (Hoods and Suits are excepted)

NOTE 1: Protection factors for respirators, as may be approved by the U.S. Bureau of Mines and/or 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 and/or 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 this part 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

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

APPENDIX B

TO

OPERATING LICENSE NO. DPR-72

FOR THE

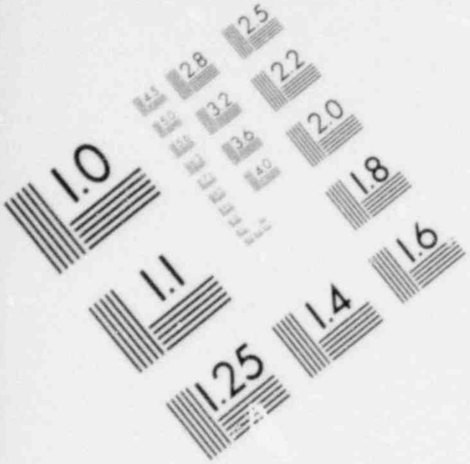
CRYSTAL RIVER UNIT 3

FLORIDA POWER CORPORATION

DOCKET NO. 50-302

DEC 3 1976

TABLE OF CONTENTS (Cont'd)



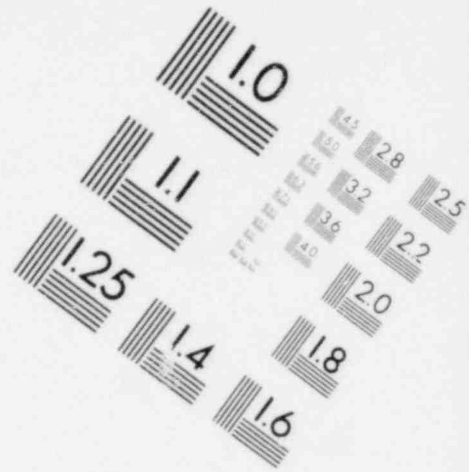
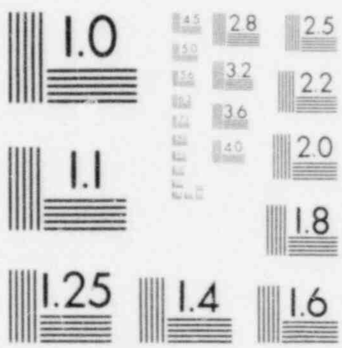


IMAGE EVALUATION
TEST TARGET (MT-3)



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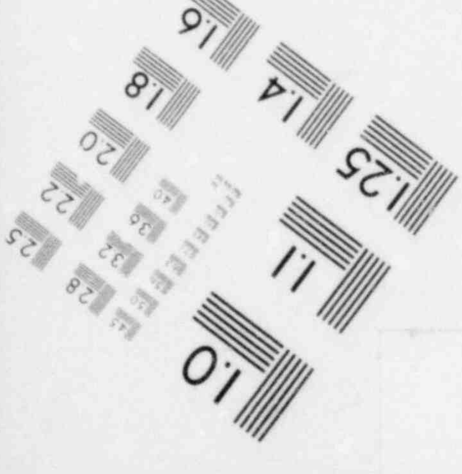
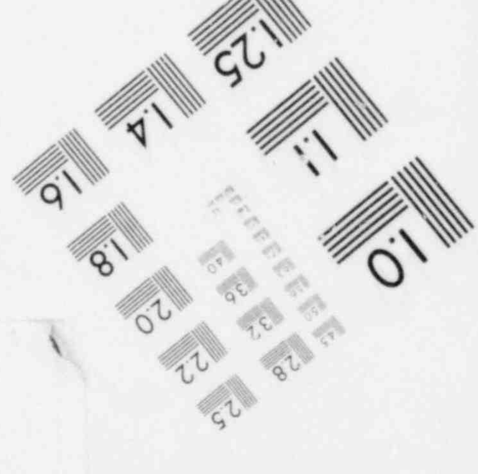
MICROCOPY RESOLUTION TEST CHART





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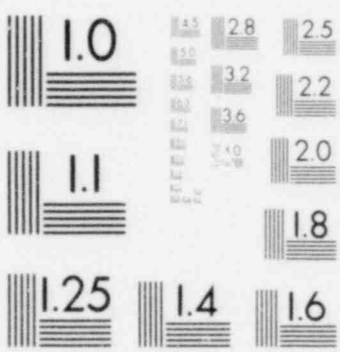

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MICROCOPY RESOLUTION TEST CHART

TABLE OF CONTENTS

List of Figures

LIST OF TABLES

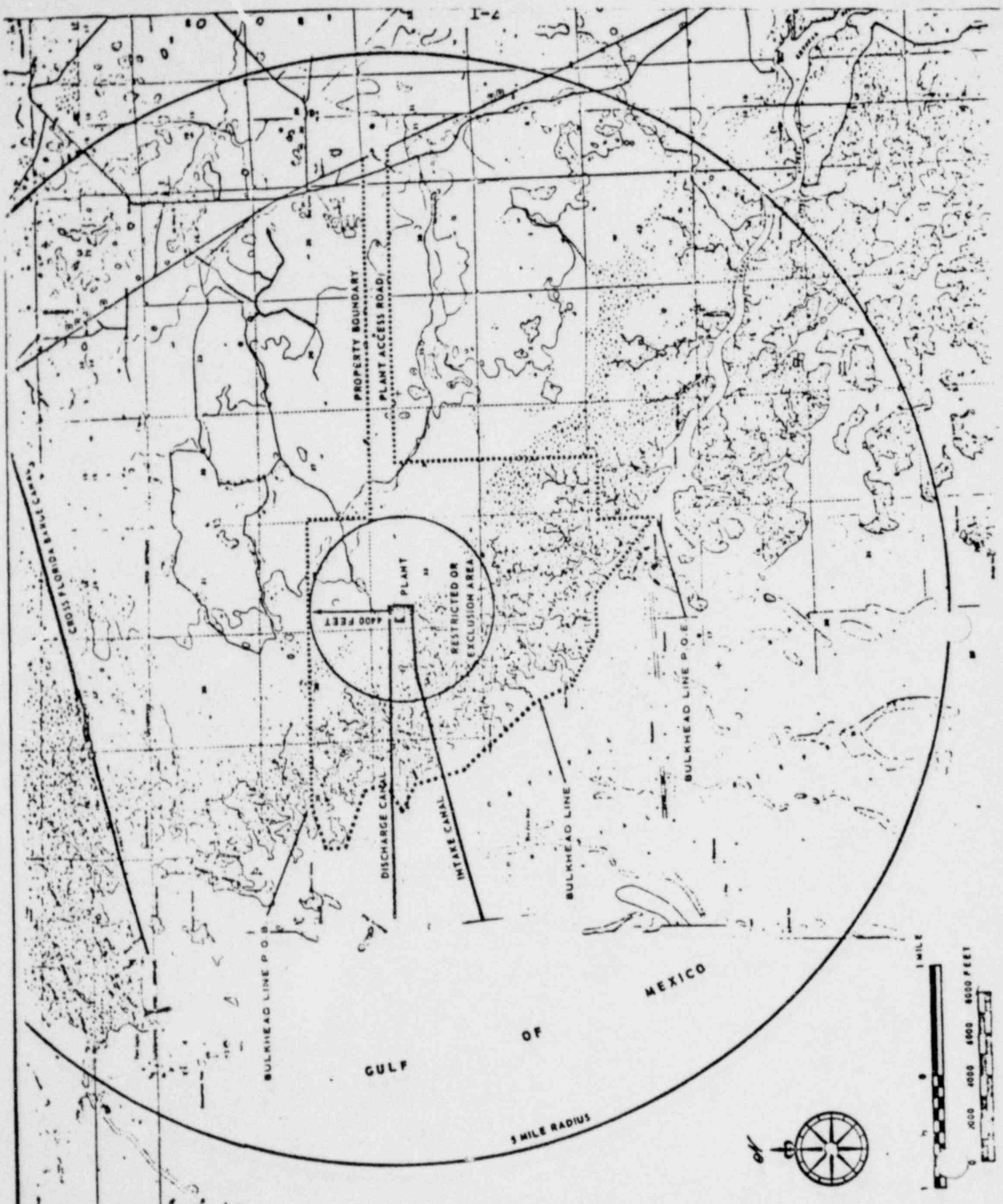


Figure 1.1-1 Point of Discharge

1.0 Definitions

The following terms are defined for uniform interpretation of the Environmental Technical Specifications for Crystal River Unit 3.

1.1 Frequency - Terms used to specify frequency are defined as follows:

One per shift - At least once per 8 hours.

Daily - At least once per 24 hours.

Weekly - At least once per 7 days.

Monthly - At least once per 31 days.

Quarterly - At least once per 92 days.

Semiannually - At least once per 6 months.

A maximum allowable extension for each surveillance requirement shall not exceed 25% of the surveillance interval.

1.2 Gross (β,γ) Analysis - Radioactivity measurements of gross beta or gross beta in conjunction with gross gamma as defined in Regulatory Guide 1.21.

1.3 Point of Discharge (POD) - The intersection of the discharge canal and the original bulkhead line as shown on Figure 1.1-1.

1.4 ΔT Across the Condenser - The average temperature difference between the inlet and outlet of Unit 3 condenser boxes.

1.5 Unit 3 Mixing Zone - The enclosed area of the discharge canal bounded by the eastern end of the canal and the cable chase from Units 1 and 2 by crossing the canal.

1.6 Emergency Need For Power - Any event causing authorized Federal officials to require or request that the Florida Power Corporation supply electricity to points within or without the State or other emergencies declared by State, County, or Municipal authorities during which an uninterrupted supply of electric power is vital to public health and safety.

1.7 Abnormal Power Operation - The operation of Crystal River Unit 3 beyond these technical specifications due to the Emergency Need for Power.

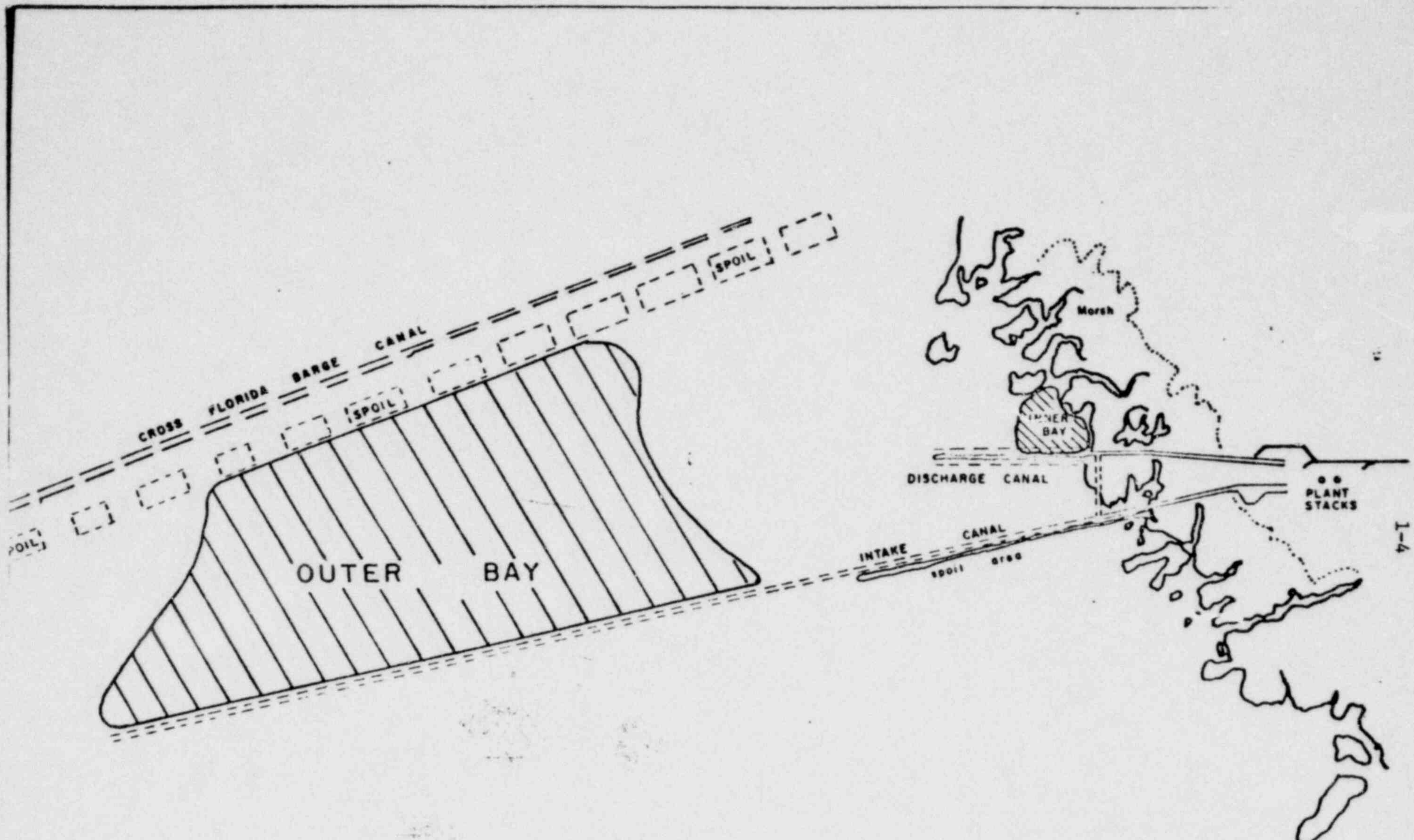


Figure 1.1-2 Inner and ter Bays

1.8 Known Radioactive Source - A calibration source which is traceable to the National Bureau of Standards radiation measurement system and is capable of reproducible geometry.

1.9 Intake Area - The intake canal and all of the water area south of the north intake dike and within two miles of the west tip of the south intake dike.

1.10 Discharge Area - The discharge canal and all of the water area north of the south discharge dike and within two miles of the north discharge dike.

1.11 Inner Bay - An area as shown in Figure 1.1-2 which is five feet or less in depth composed of a mixture of grassy bottoms, oyster associations, algal bottoms and areas of sand and mud.

1.12 Outer Bay - The outer basin as shown in Figure 1.1-2 in which the planktonic ecosystem becomes as important as the bottom ecosystems.

1.13 Channel Calibration - The adjustment, as necessary, of the channel output such that it responds with 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. Channel calibration may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

1.14 Channel Check - 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.

1.15 Channel Functional Test - 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.

1.16 Dose Equivalent I-131 - That concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in TID-14844.

to clean grass and other debris which has accumulated at the intake structure or inside the condenser water boxes. This will cause a temporary increase in the ΔT across the condenser. Because of these conditions the ΔT of 17.5°F may be exceeded for a 3 hour period with 21°F specified as a maximum limit. Monitoring by means of RTD's in the condenser inlet and outlet water boxes will provide reliable values of the ΔT across the condenser.

2.1.2 Maximum Discharge Temperature

Objective

To limit the maximum temperature of the condenser cooling water discharged from the plant to the environment during normal operation.

Specification

The temperature of the condenser cooling water at the Point of Discharge shall not exceed 103°F for a period of more than 3 consecutive hours or a maximum of 106°F unless there is an emergency need for power as defined in Section 1.

Monitoring Requirement

The temperature at the point of discharge shall be monitored once per hour during power operations of Unit 3. The temperature sensor system has a range of 30-110°F and an accuracy of ± 1/2°F. A channel check shall be performed once per month.

When the monitor is inoperative the temperature at the point of discharge shall be estimated using operating and physical data in conjunction with curves generated by an empirical analysis of the Crystal River discharge canal variables.

Bases

The effluent temperature limits during normal operations have been established to assure that the affected area within the receiving waters is minimized. Due to conditions as specified in Section 2.1.1 Bases, the condenser cooling water temperature of 103°F at the point of discharge may be exceeded for a 3 hour period with 106°F specified as a maximum limit.

2.0 LIMITING CONDITIONS FOR OPERATION

2.1 THERMAL

Objective (General)

To limit thermal stress to the aquatic ecosystem and control effluent cooling water temperature within prescribed limits which are consistent with applicable Federal and State regulations in order to minimize adverse thermal effects.

2.1.1 Maximum ΔT Across Condenser

Objective

To limit the maximum temperature rise across the condenser during normal operation at all power levels.

Specification

The temperature rise across the condenser shall not exceed 17.5°F for a period of more than 3 consecutive hours or a maximum of 21°F unless there is an emergency need for power as defined in Section 1.

Monitoring Requirement

The condenser temperature rise shall be monitored by detectors (RTD's 0-200 ± 1°F) located in the condenser inlet and outlet water boxes. The detector signal will be monitored by the control room computer. The ΔT will be alarmed at 17.5°F and at 21°F maximum.

If the RTD's or computer are inoperative during power operation above 80%, the condenser ΔT shall be determined every 2 hours ± 1 hour utilizing local temperature indicators on each water box (30-130 ± 2°F).

Bases

When Unit 3 is operated at design capacity, the intake temperature should be elevated by a value ΔT of 17.5°F. When any one shell of the two twin-shelled surface steam condensers is inoperative for maintenance or other reasons, the ΔT will rise. Each of the 4 condenser sections will require cleaning every 4 weeks, due to the buildup of marine growth or debris in the pipes and condensers. During extreme climatic conditions, especially during tropical storms, sea grass is uprooted from the Gulf of Mexico, requiring temporary shutdown of a circulator water box is being chlorinated, a sample is taken from the outlet water box, analyzed for residual chlorine (free and combined), and the results are recorded on a circular recorder. Samples are automatically changed by solenoid valves in accordance with the sequence of the chlorine generation system. This system provides a continuous record of chlorine residual in each outlet water box.

2.3.2 Corrosion Inhibitors

No chromates shall be used.

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 reasonably achievable. These releases should not result in radiation exposures in unrestricted areas greater than a few percent of natural background exposures. The concentration of effluent discharges of radioactivities 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 reasonably achievable, 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 waste:

C. The annual total quantity of noble gases above background discharged from the site should result in an air dose due to gamma radiation of less than 10 mrad, and 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.

D. The annual total quantity of all radioiodines and radioactive material in particulate forms above background from all reactors

2.2 HYDRAULIC

Not applicable.

2.3 CHEMICA'

Objective (General)

To ensure that all chemical releases from the plant are controlled and diluted so as not to adversely affect public health or the natural environment and which are consistent with applicable State and local regulations.

2.3.1 Biocides

Objective

To limit the amount and concentration of total residual chlorine in the discharge.

Specification

The concentration of total residual chlorine shall not exceed 1.8 ppm in an individual condenser outlet water box. The frequency of chlorination shall not exceed 60 minutes per water box per day and chlorination of each of the condenser circulating pipes and water boxes shall be staggered to prevent simultaneous treatment with chlorine.

Monitoring Requirement

A continuous chlorine recorder with a detection limit of 0.1 ppm shall be observed once each day on days when chlorination is performed to verify that the total residual chlorine does not exceed 1.8 ppm in the individual condenser outlet water boxes. The continuous chlorine analyzer shall be calibrated semiannually.

When the chlorine analyzer and/or recorder are inoperative, a sample shall be taken of each affected water box weekly (during chlorination) and analyzed.

Bases

Residual chlorine has a potentially detrimental effect on the estuary. The chlorine demand of the seawater at Crystal River has a range of 0.6 - 1.8 ppm. Only one of the four water boxes is chlorinated at a time. This results in water having a maximum chlorine residual of 1.8 ppm mixing with three equal volumes of water having a minimum chlorine demand of 0.6 ppm and the almost immediate chemical reduction of the remaining chlorine within the Unit 3 discharge. As each a program of action to reduce such releases to the design objective levels listed in Section 2.4, and a report of these act'ons shall be made to the USNRC in accordance with Sect. .6.2.C(1).

Liquid Waste Sampling and Monitoring Requirements

I. 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 in accordance with Section 5.6.1.B. Estimates of the sampling and analytical errors associated with each reported value shall be included.

J. 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 emitting isotope 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. All liquid wastes collected in the laundry and hot shower sump shall be transferred to the radioactive liquid waste treatment system for monitored batch release through the liquid radwaste discharge pipe.

K. Sampling and analysis of liquid radioactive waste shall be performed in accordance with Table 2.4-1. Prior to taking samples from a monitoring tank, or the laundry and hot sump , at least two tank volumes shall be circulated.

L. 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.

M. The flow rate of liquid radioactive waste shall be continuously measured and recorded during release.

N. 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. The relationship between effluent concentration and monitor readings should 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 <u>Liquid Waste Effluents</u>

<u>Specification</u>

A. The instantaneous concentration of radioactive materials released in liquid waste effluents from all reactors at the site shall not exceed the values 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.

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

B. Specification 2.4.1.B and 2.4.1. ... blish the upper limits for the release of radioactive mate... s 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.

C. Specifications 2.4.1.D and 2.4.1.E require that suitable equipment to control and monitor the releases of radioactive materials in liquid wastes are operating during any period these releases are taking place consistent with the requirements of 10 CFR Part 50, Appendix A, Design Criterion 64.

D. 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 reasonably achievable 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 reasonably achievable, 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.

E. Specification 2.4.1.G limits the amount of radioactive material that could be inadvertently released to the environment to an amount that will not exceed the Technical Specification limit.

F. In addition to limiting conditions for operation listed under Specification 2.4.1.B and 2.4.1.C, 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.

be established. Each monitor shall also have a functional test monthly and an instrument check prior to making a release.

O. The radioactivity in low power generator bleeds shall be sampled upon commencement of bleeds and every 4 hours thereafter and analyzed for gross β radioactivity concentration. Low power generator bleed with radioactivity levels in excess of Specification 2.4.1.A will be directed to the liquid radwaste system for further processing.

P. The points of release to the environment shall be monitored in accordance with Table 2.4-3.

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 reasonably achievable 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 reasonably achievable, 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 variables including defective fuel, primary system leakage, primary to secondary system leakage, and performance of the various waste treatement systems.

A. 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.

$\bar{K} = (1/Q_T) \sum_i Q_i K_i$

$\bar{L} = (1/Q_T) \sum_i Q_i L_i$

$\bar{M} = (1/Q_T) \sum_i Q_i M_i$

$\bar{N} = (1/Q_T) \sum_i Q_i N_i$

where the values of K_i, L_i, M_i and N_i are provided in Table 2.4-5 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. (Ci/sec)

A. (1) The release rate limit of noble gases from the site shall be such that

$$2.0 \left[Q_{Tv} \bar{K}_v \right] \leq 1$$

and

$$0.33 \left[Q_{Tv} (\bar{L}_v + 1.1 \bar{N}_v) \right] \leq 1$$

(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

$$3.5 \times 10^4 \, Q_v \leq 1$$

B. (1) The average release rate of noble gases from the site during any calendar quarter shall be such that

$$13 \left[Q_{Tv} \bar{N}_v \right] \leq 1$$

and

$$6.3 \left[Q_{Tv} \bar{M}_v \right] \leq 1$$

G. The sampling and monitoring requirements 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 quantities of radioactive materials released in liquid waste effluents are furnished to the Commission according to Section 5.6.1.B in conformance with Regulatory Guide 1.21. On the basis of such reports and any additional information, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate.

2.4.2 Gaseous Waste Effluents

Specification

The terms used in these Specifications are as follows:

subscripts v, refers to vent releases

i, refers to individual noble gas nuclide

(Refer to Table 2.4-5 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)

The values of $\overline{K}$, $\overline{L}$, $\overline{M}$ and $\overline{N}$ are to be determined each time isotopic anaylsis is required as delineated in Specification 2.4.2.J. Determine the following using the results of the noble gas radionuclide analysis:

(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

$$50 \left[3.5 \times 10^4 Q_v \right] > 1$$

(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 primary system waste gas holdup system the effluent monitor for the Waste Gas Storage Tanks shall be operated and set to alarm and to initiate the automatic closure of the waste gas discharge valve prior to exceeding the limits specified in 2.4.2.A above. The operability of each automatic isolation valve listed in Table 2.4-4 shall be demonstrated quarterly.

E. The maximum activity to be contained in one waste gas storage tank shall not exceed 47,000 curies (considered as Xe-133).

Gaseous Waste Sampling and Monitoring Requirements

F. 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.

G. Gaseous releases to the environment (noble gases), except from the turbine building ventilation exhaust shall be continuously monitored and recorded for gross radioactivity and the flow continuously measured and recorded. Whenever these monitors are inoperable, grab samples shall be taken and analyzed daily for gross radioactivity. If these monitors and/or recorders are inoperable for more than seven days, these releases shall be terminated.

H. During the release of gaseous wastes from the primary system waste gas holdup system, the gross activity monitor, the iodine collection device, and the particulate collection device shall be operating.

I. 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. The relationship between effluent concentration and monitor readings should

(2) The average release rate of noble gases from the site during any 12 consecutive months shall be

$$25\left[Q_{Tv}\bar{N}_v\right] \leq 1$$

and

$$13\left[Q_{Tv}\bar{M}_v\right] \leq 1$$

(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

$$13\left[3.5 \times 10^4\ Q_v\right] \leq 1$$

(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

$$25\left[3.5 \times 10^4\ Q_v\right] \leq 1$$

(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 12 consecutive months shall not exceed 4 Ci/reactor.

C. Should any of the conditions of 2.4.2.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

$$50\left[Q_{Tv}\bar{N}_v\right] > 1$$

or

$$25\left[Q_{Tv}\bar{M}_v\right] > 1$$

For Specification 2.4.2.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 2.4-5. 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 1[illegible]50 meters in the ENE 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 thyroid dose rate from radiodines 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 2.4.2.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 garden located 4800 meters in the NE direction (vent X/Q = 1.07 x 10^{-6} sec/m^3).

Specification 2.4.2.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 be established. Each monitor shall have a functional test at least monthly and instrument check at least daily.

J. Sampling and analysis of radioactive material in gaseous waste, including particulate forms and radioiodines, shall be performed in accordance with Table 2.4-2.

K. The points of release to the environment shall be monitored in accordance with Table 2.4-4.

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 reasonably achievable 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 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 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 reasonably achievable, 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, and the performance of the various waste treatment systems.

Specification 2.4.2.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.

2.4.3 Solid Waste Handling and Disposal

Specification

A. The total curie quantity and principle radionuclide composition shall be determined by measurement or estimates for 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 2.4.3 provide assurance that solid radioactive materials stored at the plant and shipped offsite are packaged in conformance with 10 CFR Part 71 and 49 CFR Parts 170-178.

the design objective annual quantity during any period of 12 consecutive months. In addition to the limiting conditions for operation of Specifications 2.4.2.A and 2.4.2.B, the reporting requirements of 2.4.2.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 2.4.2.D requires that suitable equipment to monitor and control the radioactive gaseous releases are operating during any period these releases are taking place.

Specification 2.4.2.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 on 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 semi-infinite cloud model. The calculated quantity will limit the offsite dose above background to 0.5 rem or less, consistent with Commission guidelines.

The sampling and monitoring requirements given under Specification 2.4.2 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.

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 2.4-4.

Specification 2.4.2.G 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.

Table 2.4-1 (Continued)

NOTES:

(1) A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged.

(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) 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.

(4) The power level and cleanup or purification flow rate at the sample time shall also be reported.

(5) For dissolved noble gases in water, assume an MPC of 4×10^{-5} μCi/ml of water.

Table 2.4-1

RADIOACTIVE LIQUID SAMPLING AND ANALYSIS

	Liquid Source	Sampling Frequency & Analysis	Type of Activity Analysis	Detectable Concentrations (μCi/ml)[3]
A.	Batch Releases to the Environment	Each Batch (Grab Sample)	Principal Gamma Emitters	5×10^{-7} (2)
			Ba-La-140, I-131	10^{-6}
		One Batch/Month (Grab Sample)	Dissolved Gases (5)	10^{-5}
		Monthly Composite[1] (From Grab Samples)	H-3	10^{-5}
			Gross α	10^{-7}
			Sr-90, Sr-89	10×10^{-8}
B.	Primary Coolant	At least Biweekly[4]	I-131, I-133	10^{-6}
C.	Low Power Generator Bleeds	Commencement of Bleed (Grab Sample)	Gross β	5×10^{-7}
		Every 4 hrs during Bleed	Gross β	5×10^{-7}

Table 2.4-2 (Continued)

NOTES:

(1) The above detectability limits for activity 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) Analyses shall be performed following each refueling, startup or similar operational occurrence which could alter the mixture of radionuclides.

(3) 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.

(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.

Table 2.4-3

PWR-LIQUID WASTE SYSTEM

LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLES REQUIRED BY TECHNICAL SPECIFICATIONS

Process Stream or Release Point	Radiation Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	Gross Activity	I	Dissolved Gases	Alpha	H-3	Isotopic Analysis	High Liquid Level Alarm
Evaporator Condensate Storage Tanks (A & B)				X		X	X	X	X	X	X
Laundry & Shower Sump				X		X	X	X	X	X	X
Primary Coolant System				X		X					
Liquid Radwaste Discharge Pipe	X	X	X		X						
Outdoor Storage Tanks (potentially radioactive)				X	X						
Condensate Storage & Secondary Neutralizer Tank				X	X					X*	X
Component Cooling Systems	X		X		X						
Turbine Building Sumps (Floor Drains)				X	X					X*	X
Nuclear Service Area Sump				X	X					X*	

*Lab analysis capability

Table 2.4

RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS FREQUENCY

	Gaseous Source	Sampling and Analysis Frequency	Type of Activity Analysis	Detectable Concentrations (μCi/ml)(1)
A.	Waste Gas Decay	Each Tank (Grab Sample)	Principal Gamma Emitters	10^{-4} (3)
	Tank Releases	Release	H-3	10^{-6}
B.	Containment Purge	Each Purge	Principal Gamma Emitters	10^{-4} (2)
	Releases	(Grab Sample)	H-3	10^{-6}
C.	Condenser Air Ejector	Weekly (Grab Sample)	Principal Gamma Emitters	10^{-4} (2) (3)
		Monthly (Grab Sample)	H-3	10^{-6}
D.	Environmental Release	Weekly (Gas Grab Sample)	Prinicpal Gamma Emitters	10^{-4} (2)(3)
	Points	Monthly (Gas Grab Sample) (On Line RMS)	H-3	10^{-6}
		Weekly (Charcoal Filter) (On Line RMS)	I-131	10^{-12}
		Weekly (Charcoal Filter) (On Line RMS)	I-133, I-135	10^{-10}
		Weekly (Particulate Filter)	Principal Gamma Emitters (Ba-La-140, I-131 and others)	10^{-11}
		Monthly Composite(4) (Particulate Filters)	Gross α	10^{-11}
		Monthly Composite(4) (Particulate Filters)	Sr 89, Sr 90	10^{-11}

Table 2.4-4

PWR-GASEOUS WASTE SYSTEM

LOCATION OF PROCESS AND EFFLUENT MONITORS AND SAMPLES REQUIRED BY TECHNICAL SPECIFICATIONS

Process Stream or Release Point	Radiation Alarm	Auto Control to Isolation Valve	Continuous Monitor	Grab Sample Station	Measurement Capabilities				
					NC	I	Part	H-3	Alpha
Process Stream									
Waste Gase Decay Tank	X	X	RMA-11	X	X	X	X	X	X
Condenser Vacuum Pump Exhaust	X		RMA-12	X	X	X	X	X	X
Building Ventilation Systems									
Reactor Building Purge Exhaust Duct [Whenever there is flow]	X	X	RMA-1	X	X	X	X	X	X
Auxiliary Building and Fuel Handling Building Exhaust Duct*	X	X	RMA-2	X	X	X	X	X	X

*This exhaust includes the radwaste area.

Table 2.4-5

GAMMA AND BETA DOSE FACTORS FOR CRYSTAL RIVER UNIT 3

$x/Q = 1.46 \times 10^{-6}$ sec/m^3 at 1450 meters, ENE

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	7.0×10^{-5}	0	0.92	0.035
Kr-85m	0.80	2.1	2.9	0.84
Kr-85	0.0096	2.0	2.8	0.010
Kr-87	2.5	14	15	2.6
Kr-88	6.1	3.5	4.3	6.4
Kr-89	2.79	15	15	0.83
Xe-131m	2.28	0.69	1.6	0.35
Xe-133m	0.22	1.5	2.2	0.29
Xe-133	0.26	0.55	1.5	0.31
Xe-135m	1.2	1.0	1.1	1.1
Xe-135	1.2	2.7	3.6	1.3
Xe-137	0.12	18	19	0.12
Xe-138	2.4	6.0	6.9	2.5

In this particular system the final stabilized level is higher than the initial level and is only obtained after a period of stabiliztion and after going through a suppressed level following the initial perturbation. The recognition of this type of potential response is obviously important in considering any surveillance program.

The models of the systems involved at Crystal River along with the data available indicate that the approximate time to stabilization should not exceed one year. Therefore, the time frame for the intensive surveillance program elements allows one year of monitoring to determine the transient response that the systems are experiencing. An additional year of monitoring is required to indicate the new stabilized level. If the second year's data indicate that the systems have not approached stabilization, the monitoring will be extended for an additional year. It is anticipated that the intensive surveillance program elements should not be necessary beyond three years.

The areas in which intensive monitoring will be performed as indicated or until stabilization occurs include the following program elements: (1) Thermal plume model verification, (2) Benthos in discharge area, (3) Marsh grasses, and (4) Impingement on intake screens.

In addition to the short-term intensive surveillance program elements designed to determine how the systems have responded to the perturbations, an on-going program element designed to obtain a diagnostic view of the condition of the environment will be continued during the operational life of the plant. This indicator program element consists of a number of simple measurements which will detect any major changes in the system. A second long-term program element involves chemical-industrial waste water monitoring.

3.1.1 Benthos in Discharge Area

Objective

To determine the ecol[illegible]l condition of the benthic system in the area directly affecte[illegible] the thermal plume.

Specification

Operational monitoring of productivity, respiration, diversity and biomass of the benthic system in the area adjacent to and north of the discharge canal shall be measured on a quarterly basis until the system has approached stabilization. Samples shall be taken by methods implied in the preoperational studies including harvesting quadrats, by sediment cores, and by venturi pumps. The

3.0 ENVIRONMENTAL SURVEILLANCE

3.1 NONRADIOLOGICAL SURVEILLANCE

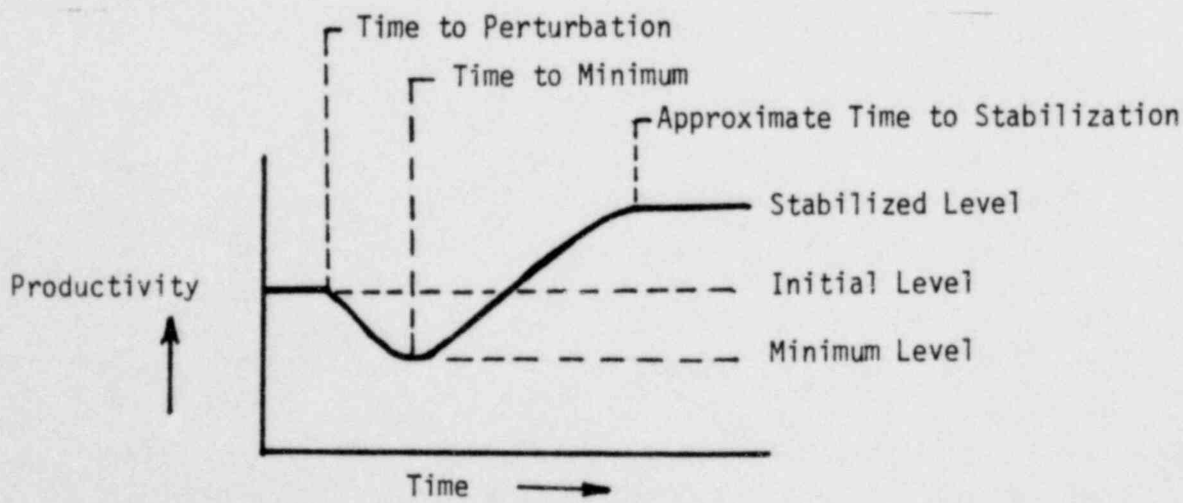
Study Plan

The estuary has been exposed to the influence of the operation of Units 1 and 2 for approximately seven (7) years. During this time, the systems in the area have adapted to this influence. A preoperational surveillance program was designed to determine the exact nature of the new stabilized conditions relative to control areas adjacent to the plant site. This surveillance consisted of system modeling with measurements of biomass, productivity, respiration and diversity in all major compartments. The information derived will serve as a baseline for comparison with the data taken after Unit 3 becomes operational.

The operational surveillance program is designed to determine any significant environmental effects of the operation of the power plant, particulary unpredicted and catastrophic changes. The program consists of 4 short-term intensive surveillance program elements and 2 long-term program elements.

A period of adjustment of the ecosystem is expected concurrent with Crystal River Unit 3's initial operation. This will be a localized perturbation limited to a portion of the inner bay associated with the higher water velocity as well as the temperature increase resulting from the condenser discharge.

Any ecosystem which experiences a change in its environment will undergo a period of adaptation unless catastorphic conditions occur. With the small changes anticipated with the addition of Unit 3, no catastrophic effects are expected. However, any changes in the environmental conditions of a system will normally cause it to oscillate. An example of the oscillation of a hypothetical system's productivity is shown below.



Bases

The metabolism of the marsh grass is expected to increase with increasing temperature. Any decrease indicates a breakdown of structure. If any of these parameters changes beyond 2σ (two standard deviation) of that measured during preoperational monitoring, the system should be investigated for catastrophic results.

3.1.3 Impingement on Intake Screens

Objective

To determine the quantity of impinged fish and shellfish on the intake screens to compare with preoperational data.

Specification

The fish and shellfish collected in the trash racks adjacent to the intake screens of Units 1 and 2 and Unit 3 will be sampled for 24 consecutive hours once weekly. This program shall be conducted for one year after operation of Unit 3 begins. This program may be terminated after one year period with staff's approval. Samples shall be sorted according to species, length, and wet weight. The screen-wash racks shall be monitored visually daily to determine any abnormal catches.

Reporting Requirement

Results of the data gathered in this program element shall be reported in accordance with Section 5.6.1. Any daily sample with fish and shellfish biomass greater than 50 kg shall be reported as specified in Section 5.6.2.

Bases

Preoperational data indicate that the average normal expected catch is approximately 20 kg per day for Unit No. 3.

3.1.4 General Ecological Survey

Objective

To detect changes which might occur and would be used to indicate areas requiring more detailed investigation.

Specifications

A series of measurements shall be carried out during the operational life of the plant to indicate the general condition of the environment. The areas to be monitored are:

number, frequency and location of samples to be taken shall be determined from a statistical analysis of the research presently being conducted in this area. Samples shall be stratified by macrophyte dominance. Productivity and respiration of the system shall be determined by the methods currently employed in the modeling work.

Reporting Requirement

Results of the data gathered in this program element shall be reported in accordance with Section 5.6.1. In the event that any parameter measured changes beyond two standard deviations of the value measured in the preoperational monitoring program, a report shall be submitted as specified in Section 5.6.2.

Bases

In the discharge area adjacent to the canal, the productivity, respiration and biomass should increase due to an increased temperature of the cooling water. If any of these parameters changes beyond 2σ (two standard deviation) of that measured during preoperational monitoring, the system should be investigated for catastrophic results.

.1.2 Marsh Grass

Objective

To determine the ecological condition of the salt marsh adjacent to the discharge area.

Specification

The biomass, productivity, and respiration of the salt marsh shall be measured on a quarterly basis after plant operation begins until the system has approached stabilization. Quadrats shall be harvested to determine biomass and productivity.

Reporting Requirement

Results of the data gathered in this program element shall be reported in accordance with Section 5.6.1. In the event that any parameter measured changes beyond 2σ (two standard deviation) of the value measured in the preoperational monitoring program, a report shall be submitted as specified in Section 5.6.2.

Reporting Requirements

Reports shall be furnished in accordance with Section 5.6.1.

Basis

Monitoring is performed to assure that all chemical-industrial waste water is controlled so as to not adversely affect public health or the natural aquatic environment.

3.2 RADIOLOGICAL ENVIRONMENTAL MONITORING

Objective

The radiological environmental monitoring program will provide information which can be used to assist in assessing the type and quantity of radiation exposure in unrestricted areas resulting from plant operation.

Background

Preoperational radiological environmental monitoring programs, to establish baseline environmental concentration values, were initiated in mid-1970. One program was operated by the State of Florida Department of Health and Rehabilitative Services; another program was operated by the University of Florida.

A summary of the preoperational surveillance results is shown in Table 3.2-1. This summary includes median values of the observed environmental concentrations and 95 percentile values (i.e., values which exceed 95 percent of all the comparable measured values). These values will be taken as the preoperational baseline concentrations. In some cases the values listed are smaller than the Lower Limit of Detection (LLD).

The 95 percentile values indicate the random frequency of high measured values during the operation of the plant contributes negligibly to the environmental radioactivity. These 95 percentile values will be used during operation to assess the probability that any observed high concentration value is due to random fluctuations in measurements rather than to a true increase in environmental concentrations.

Specification (Program)

Environmental media which are sampled and analyzed for radioactivity are shown by the two diagrams on Figure 3.2.-1. Each box in the diagrams contains the name of an environmental media which is sampled. The upper diagram shows the critical pathways; the lower diagram shows the other monitored pathways.

a. Outer bay (plankton-dominated area). The percent of saturation of oxygen will be measured at dusk and dawn of consecutive days, twice monthly. In addition, a semiannual sample of the zooplankton shall be taken with 202 μ zooplankton net. Taxsonomic composition abundance and total biomass will be determined.

b. Canals. The percent of saturation of oxygen shall be measured at dusk and dawn twice monthly at the Point of Discharge.

c. Inner bay. Quarterly tows by a man in a glass-bottomed boat or alternative method approved by the NRC staff to observe the general condition and percent cover of the sea grasses shall be made. These tows shall be along a transect along a radial from point of discharge.

d. Oyster reefs. Counts of the organism within a quadrat on the reef shall be made quarterly.

e. Marsh grasses. Stem counts of grass within a quadrat will be made quarterly. This measurement shall be correlated with biomass. In addition, the number of crab holes within a quadrant shall be observed as a biomass indicator.

Reporting Requirement

Results of the data gathered in this program element will be reported in accordance with Section 5.6.1.

Bases

The parameters to be measured were chosen to indicate general trends in the conditions of the environment and will be used to indicate areas where further investigations may be warranted if significant changes are detected.

3.1.5 Chemical-Industrial Waste Water Treatment System

Objective

To monitor the Chemical-Industrial Waste Water Ponds and surrounding environs.

Specification

Representative samples shall be obtained and analyzed once per month from test wells 1, 4, and 5, the discharge canal directly to the North of the ponds, and the ponds themselves. (See Figure 3.1-1) Samples are analyzed for pH Nitrate (NO_3), Sulfate (SO_4), Phosphate (PO_4), Iron (Fe), dissolved solids, Copper (Cu), and Zinc (Zn).

The operational radiological monitoring program shall consist of a continuation of the preoperational program of measurements of radioactivity in environmental media which is outlined in Table 3.2-2.

The critical pathway monitoring program which is included in Table 3.2-2 is also shown in Table 3.2-3. Sample station locations are described on Table 3.2-4 and shown on maps on Figures 3.2-2 and 3.2-3. Lower Limit of Detection (LLD) values are given on Table 3.2-5.

Deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, vandalism, seasonal unavailability or to malfunction of automatic sampling equipment. If the latter, every 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.

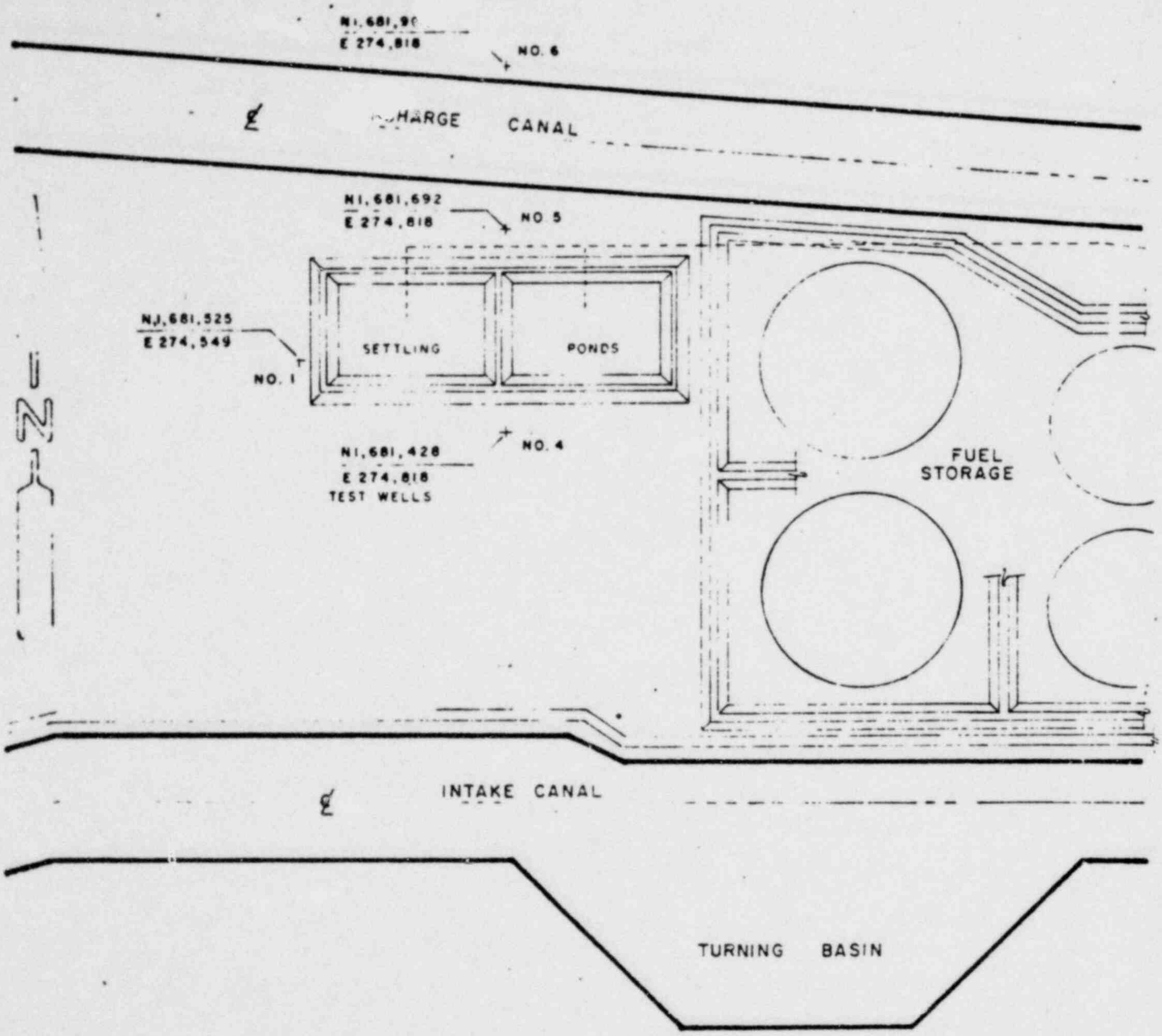


Figure 3.1-1 Settling Ponds and Test Well Locations

TABLE 3.2-1

SUMMARY OF PREOPERATIONAL ENVIRONMENTAL
SURVEILLANCE RESULTS 1971 - 1974 (Cont'd.)

Environmental Media	Nuclide	Preoperational Concentrations: Median Value	95 Percentile Value
Sea Water	H-3	71. pCi/l	87 pCi/l
	Ba-140	(a)	11. pCi/kg
	Cs-137	9(a)	10. pCi/kg
	Zn-65	(a)	7. pCi/kg
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	K-40	0.18 g/kg	0.44 g/kg
	Ra-226	(a)	600. p Ci/kg
	Th-232	(a)	7. pCi/kg
	Zr-95	(a)	(a)
	Ru-106	(a)	(a)
	Cs-134	(a)	(a)
Air	Gross β	0.029 pCi/kg	0.120 pCi/kg
	Ba-140	(a)	0.016 pCi/m^3
	Cs-137	(a)	0.013 pCi/m^3
	Zn-65	(a)	(a)
	Mn-54	(a)	(a)
	I-131	(a)	0.004 pCi/m^3
	K-40	(a)	(a)
	Ra-226	(a)	0.241 pCi/m^3
	Th-232	(a)	0.008 pCi/m^3
	Zr-95	0.003 pCi/m^3	0.043 pCi/m^3
	Ru-106	0.025 pCi/m^3	0.216 pCi/m^3
	Ce-144	0.003 pCi/m^3	0.172 pCi/m^3
Milk	Sr-90	4. pCi/l	6.pCi/l
	Cs-134	(a)	(a)
	I-131	(a)	(a)
	Ba-140	(a)	(a)
	Co-58	(a)	(a)
	Co-60	(a)	(a)
	Mn-54	(a)	(a)
	Zr-95	(a)	(a)
	Cs-137	16 pCi/l	22 pCi/l

(a) See pg 3-10.

TABLE 3.2.1

SUMMARY OF PREOPERATIONAL ENVIRONMENTAL
SURVEILLANCE RESULTS 1971 - 1974

Environmental Media	Nuclide	Preoperational Concentrations: Median Value	Preoperational Concentrations: 95 Percentil Value
Water, Potable	Gross β	(a)	19. pCi/l
	H-3	(a)	(a)
	Co-58	(a)	(a)
	Co-60	(a)	(a)
	Ba-140	(a)	(a)
	Cs-134	(a)	(a)
	Cs-137	(a)	(a)
	Zn-65	(a)	(a)
	Mn-54	(a)	(a)
	I-131	(a)	(a)
Water, Surface	Gross β	(a)	19. pCi/l
	H-3	(a)	(a)
	Co-58	(a)	(a)
	Co-60	(a)	(a)
	Ba-140	(a)	(a)
	Cs-134	(a)	(a)
	Cs-137	(a)	(a)
	Zn-65	(a)	(a)
	Mn-54	(a)	(a)
	I-131	(a)	(a)
Water, Precipitation	Gross β	(a)	19. pCi/l
	H-3	(a)	(a)
	Co-58	(a)	(a)
	Co-60	(a)	(a)
	Ba-140	(a)	(a)
	Cs-134	(a)	(a)
	Cs-137	(a)	(a)
	Zn-54	(a)	(a)
	I-131	(a)	(a)

[a] The median value is less than the Lower Limit of Detection (LLD); the median value is assumed to be LLD (Table 3.2-5, A and B).

Table 3.2-1

SUMMARY OF PREOPERATIONAL ENVIRONMENTAL SURVEILLANCE RESULTS 1971 - 1974 (Cont'd)

Environmental Media	Nuclide	Preoperational Concentrations: Median Value	95 Percentile Value
Oyster Meat	Ba-140	(a)	(a)
	Cs-137	(a)	(a)
	Zn-65	(a)	33. pCi/kg
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	K-40	(a)	2.2 g/kg
	Ra-226	(a)	534. pCi/kg
	Th-232	(a)	(a)
	Zr-95	(a)	(a)
	Ru-106	(a)	82. pCi/kg
Blue Crab	Ba-140	(a)	55. pCi/kg
	Cs-137	(a)	75. pCi/kg
	Zn-65	(a)	127. pCi/kg
	Mn-54	(a)	24. pCi/kg
	I-131	(a)	(a)
	K-40	1.7 g/kg	2.4 g/kg
	Ra-226	1325. pCi/kg	3600. pCi/kg
	Th-232	92. pCi/kg	170. pCi/kg
	Zr-95	(a)	13. pCi/kg
	Ru-106	(a)	(a)
	Cs-134	(a)	(a)
Herbivorous Fish	Ba-140	(a)	50. pCi/kg
	Cs-137	(a)	110. pCi/kg
	Zn-65	(a)	63. pCi/kg
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	K-40	2.6 g/kg	3.7 g/kg
	Ra-226	960. pCi/kg	3100. pCi/kg
	Th-232	(a)	84. pCi/kg
	Zr-95	(a)	9. pCi/kg
	Ru-106	(a)	90. pCi/kg
	Cs-134	(a)	(a)

[a] See page 3-10.

Table 3.2-1

SUMMARY OF PREOPERATIONAL ENVIRONMENTAL
SURVEILLANCE RESULTS 1971 - 1974 (Cont'd)

Environmental Media	Nuclide	Preoperational Concentrations: Median Value	Preoperational Concentrations: 95 Percentile Value
Soil	Ba-140	(a)	(a)
	Cs-137	270. pCi/kg	1100. pCi/kg
	Zn-65	(a)	(a)
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	K-40	713 pCi/kg	1482 pCi/kg
	Ra-226	(a)	2200. pCi/kg
	Th-232	(a)	300. pCi/kg
	Zr-95	40. pCi/kg	150. pCi/kg
	Ru-106	0. pCi/kg	330. pCi/kg
Small Terrestrial Animals	Ba-140	(a)	(a)
	Cs-137	(a)	80. pCi/kg
	Zn-65	(a)	160. pCi/kg
	Mn-54	(a)	(a)
	I-131	(a)	100. pCi/kg
	K-40	3.17 g/kg	4.28 g/kg
	Ra-226	(a)	720. pCi/kg
	Th-232	(a)	(a)
	Zr-95	(a)	70. pCi/kg
	Ru-106	(a)	(a)
Total Deposition	Ba-140	3 pCi/m^2-day	22 pCi/m^2-day
	Cs-137	(a)	6 pCi/m^2-day
	Zn-65	(a)	7 pCi/m^2-day
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	K-40	(a)	.03 g/m^2-day
	Ra-226	20 pCi/m^2-day	508. pCi/m^2-day
	Th-232	(a)	6. pCi/m^2-day
	Zr-95	(a)	13. pCi/m^2-day
	Ru-106	(a)	152. pCi/m^2-day

[a] See page 3-10.

Table 3.2-1

SUMMARY OF PREOPERATIONAL ENVIRONMENTAL SURVEILLANCE RESULTS 1971 - 1974 (Cont'd)

Environmental Media	Nuclide	Preoperational Concentrations: Median Value	Preoperational Concentrations: 95 Percentile Value
Vegetation	K-40	.69 g/kg	2.9 g/kg
	Ra-226	(a)	2363. pCi/kg
	Th-232	(a)	120. pCi/kg
	Ba-140	26. pCi/kg	253. pCi/kg
	Zr-95	(a)	31. pCi/kg
	Cs-137	1363. pCi/kg	5416. pCi/kg
	Zn-65	(a)	589. pCi/kg
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	Ru-106	(a)	(a)
Aquatic Plants and Grasses (Including Algae and Plankton)	Ba-140	(a)	75. pCi/kg
	Cs-137	(a)	181. pCi/kg
	Zn-65	(a)	156. pCi/kg
	Mn-54	(a)	43. pCi/kg
	I-131	(a)	37. pCi/kg
	K-40	1.8 g/kg	15. g/kg
	Ra-226	624. pCi/kg	3300. pCi/kg
	Th-232	(a)	280. pCi/kg
	Ru-106	(a)	360. pCi/kg
	Zr-95	18. pCi/kg	157. pCi/kg
External Radiation	All	62. mrem/yr	77. mrem/yr
Ocean Sediment	Ba-140	(a)	(a)
	Cs-137	(a)	250. pCi/kg
	Zn-65	(a)	(a)
	Mn-54	(a)	19. pCi/kg
	I-131	(a)	34. pCi/kg
	K-40	.31 g/kg	1.2 g/kg
	Ra-226	2900. pCi/kg	10000. pCi/kg
	Th-232	90. pCi/kg	300. pCi/kg
	Ru-106	190. pCi/kg	690. pCi/kg
	Zr-95	12. pci/kg	40. pCi/kg
	Cs-134	(a)	(a)

(a) See page 3-10.

Table 3.2-1

SUMMARY OF PREOPERATIONAL ENVIRONMENTAL SURVEILLANCE RESULTS 1971 - 1974 (Cont'd)

Environmental Media	Nuclide	Preoperational Concentrations: Median Value	95 Percentile Value
Carnivorous Fish	Ba-140	(a)	72. pCi/kg
	Cs-137	(a)	43. pCi/kg
	Zn-65	(a)	99. pCi/kg
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	K-40	2.8 g/kg	4.6 g/kg
	Ra-226	335. pCi/kg	2400. pCi/kg
	Th-232	(a)	92. pCi/kg
	Zr-95	(a)	12. pCi/kg
	Ru-106	(a)	(a)
	Cs-134	(a)	(a)
Shrimp	Ba-140	(a)	(a)
	Cs-137	(a)	37. pCi/kg
	Zn-65	(a)	(a)
	Mn-54	(a)	(a)
	I-131	(a)	(a)
	K-40	1.1 g/kg	3. g/kg
	Ra-226	(a)	(a)
	Th-232	(a)	36. pCi/kg
	Zr-95	(a)	(a)
	Ru-106	(a)	(a)
Food Crops (Oranges)	Co-58	(a)	(a)
	Co-60	(a)	(a)
	Ba-140	(a)	(a)
	Sr-90	105. pCi/kg	130. pCi/kg
	Cs-134	(a)	(a)
	Cs-137	(a)	(a)
	Zn-65	(a)	(a)
	Mn-54	(a)	(a)
	I-131	(a)	(a)

[a] See page 3-10.

Table 3.2-2

GENERAL PATHWAY RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (cont'd)

Pathway	Sample Type	Sample Stations	Sampling/Collection and Analysis Frequency	Analysis Routine
Potable Water	Water	C07, C10, C18	Quarterly	Extended γ-Spectral Analysis, H-3
Shoreline External Sediment	Bottom Sediments	C01, C09, C14H C14M, C14G	Semiannual	Extended γ-Spectral Analysis; Sr-90,
Sea Food Chain	Marine Plants (Marco Algae and submerged Sea Plants)	C29, C30	Semiannual	γ-Spectral Analysis Sr-89, 90
Ingestion Crab	Crab	C29, C30	Semiannual	γ-Spectral Analysis on edible portions
Ingestion Fish	Carnivorous Fish	C29, C30	Semiannual or in Season	γ-Spectral Analysis on edible portions
Ingestion Fish	Herbivorous Fish	C29, C30	Semiannual or in Season	γ-Spectral Analysis on edible portions
Ingestion Oysters	Oysters	C29, C30	Semiannual	γ-Spectral Analysis on edible portions
Ingestion Shrimp	Shrimp	C27	Semiannual	γ-Spectral Analysis on edible portions
Ingestion Milk	Milk	C47, C49	Monthly	I-131 Analysis: Extended γ-Spectral Analysis, Sr-89, 90

Table 3.2-2

GENERAL PATHWAY RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Pathway	Sample Type	Sample Stations	Sampling/Collection and Analysis Frequency	Analysis Routine
Air Submersion	External Radiation	C04, C07, C09 C18, C26, C40 C41, C43, C14H, C14M, C14G, C46	Continuous/Quarterly	TLD
Air Inhalation	Air	C04, C07, C18 C26, C40, C41 C46	Continuous/Weekly	Gross B and I-131 weekly, γ-Spectral on quarterly composite, Sr-89, 90 of quarterly composite
Precipitation	Total Deposition	C04, C26, C40	Continuous/Monthly	H-3, γ-Spectral Analysis
Sea Water	Water	C01, C09, C13 --- C14H, C14M, C14G --	Monthly Monthly	γ-Spectral Analysis, composite for Sr-89, 90, H-3 quarterly
River Water	Water	C15	Quarterly	H-3, γ-Spectral Analysis
Ground Water	Water	C40	Semiannual	γ-Spectral Analysis H-3

Table 3.2-3

CRITICAL PATHWAY RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Pathway	Sample Type	Sample Stations	Sampling/Collection Frequency	Analysis Routine	*Critical Radionuclides
Air Submersion	External Radiation	C14H, C14M C14G	Continuous/Quarterly	TLD	XE-133, Kr-88
Air Inhalation	Air	C41	Continuous/Weekly	I-131 Analysis	I-131
Ingestion Milk	Milk	C49	Monthly	I-131 Analysis	I-131
Ingestion Crab	Crab	C29	Semiannual	γ-Analysis on edible portions	Cs-134, Cs-137 I-131
Ingestion Fish	Fish	C29	Semiannual in Season	γ-Analysis on edible portions	Cs-134, Cs-137 I-131
Shoreline External Sediment	Bottom Sediments	C14H, C14M C14G	Semiannual	γ-Analysis	Cs-134, Cs-137
Boating, Swimming External	Sea Water	C14G	Monthly	γ-Analysis	Cs-134, Cs-137 I-131
Ingestion Vegetable	Green Leafy Vegetables	C48	Semiannual	γ-Analysis	I-131

*Nuclides which contribute at least 70% of dose in the pathway.

Table 3.2-2

GENERAL PATHWAY RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM (cont'd)

Pathway	Sample Type	Sample Stations	Sampling/Collection and Analysis Frequency	Analysis Routine
Ingestion Animals	Small Terrestrial Animals	C45	Semiannual	γ-Spectral Analysis
Food Chain	Vegetation (grasses)	C05, C40, C41	Semiannual	γ-Spectral Analysis
Ingestion Food Crops	Food Crops (Citrus)	C19	Annual or at harvest	γ-Spectral Analysis
Ingestion Food Crops	Food Crops (Watermelon)	C04	Annual or at harvest	γ-Spectral Analysis
Food Chain	Soil	C04, C07, C18 C26, C40, C41, C46	First year, then every Third	Extended γ-Spectral Analysis
Food Chain	Meat	C50	Semiannual	γ-Spectral Analysis
Food Chain	Poultry & Eggs	C51	Semiannual	γ-Spectral Analysis
Food Chain	Green Leafy Vegetables	C48, C47	Semiannual	γ-Spectral Analysis Sr-90

Note: All Sampling Stations, except those sampling stations listed in Table 3.2-3, are background (control) stations.

Table 3.2-4

OPERATIONAL SAMPLE STATION LOCATIONS (cont'd)

Station Number	Location	Distance from plant (miles)	Direction from plant
C27	Ralston Purina Research Facility between Intake and Discharge Canals	0.6	WNW
C29	Discharge Area	2.0	W
C3C[(a)]	Intake Area	3.6	WSW
C40	On site near N.E. Boundary at excavated pond and pump station	3.5	E
C41	Onsite meteorological	0.4	SSW
C43	S.E. corner Hollins Wood	4.7	ESE
C45	Between C40 and C46	NA	NA
C46	North Pump Station	0.4	N
C47	Univ. of Fla.,Gainesville	52.0	NNE
C48	To be determined from Anual Vegetable Garden Census		
C49	To be determined from Semiannual Milk Animal Census		
C50	To be within 10 miles of the Plant		
C51	To be within 10 miles of the Plant		

(a) This station is considered as background station for marine sampling.

Table 3.2-4

OPERATIONAL SAMPLE STATION LOCATIONS

Station Number	Location	Distance from plant (miles)	Direction from plant
C01	Levy County Park West end State Road 40	4.9	NW
C04	State Park Old Dam on River near road intersection	6.3	ENE
C05	Entrance to Hollins Wood Ranch	3.8	ENE
C07	Crystal River Public Water Plant	7.5	ESE
C09	Citrus County Park West End State Road 44	3.2	S
C10	Indian Waters Public Water Supply	5.9	ESE
C13	Mouth of Intake Canal	3.4	WSW
C14H	Head of Discharge Canal	0.1	NW
C14M	Midway out Discharge Canal	1.5	W
C14G	Gulf of Mexico end of Discharge Canal	2.8	W
C15	Withlacoochee River Yankeetown Dock Isaac Walton Lodge	5.0	N
C18	Yankeetown City Well	5.2	N
C19	NW Corner State Road 488 and State Road 495	8.5	ENE
C26	FPC Substations on State Rd. 491 between Beverly Hills and Holder	16.6	E

Table 3.2-5B

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

LLD Analysis	Water [pCi/kg]	Airborne Particulate [pCi/m^3]	Fish, Meat, and Poultry [pCi/kg wet]	Milk [pCi/l]	Vegetation [pCi/kg wet]	Soil (pCi/kg dry]	Total Deposition pCi/m^2/day
Gross Beta(b)	1	0.01	NA	NA	NA	NA	NA
H-3(b)	200	NA	NA	NA	NA	NA	NA
K-40(c)	51	0.11	394	NA	248	328	NA
Mn-54	15	0.01	32	NA	21	27	NA
Co-58	17	NA	60	NA	24	NA	NA
Fe-59	30	NA	260	NA	NA	NA	NA
Co-60	17	NA	60	NA	24	NA	NA
Zn-65	30	0.02	66	NA	42	55	NA
Sr-89	5	0.0019	NA	5	11	NA	NA
Sr-90	2	0.0008	24	2	4	135	NA
Zr-95	14	0.01	31	NA	20	26	NA
Ru-106	84	0.05	184	NA	118	153	NA
I-131	17	0.01	38	10	24	32	NA
Cs-134	17	0.01	60	15	24	50	NA
Cs-137	17	0.01	37	10	24	50	NA
Ba-140	17	0.01	39	10	24	32	NA
Ce-144	NA	0.06	NA	NA	NA	NA	NA
Ra-226	57	0.04	200	NA	80	167	NA
Th-232	28	0.02	100	NA	40	83	NA

(a) per State of Florida Department of Health and Rehabilitation Services.
(b) Gross Beta and H-3 are reported in pCi/l.
(c) K-40 is reported in g/kg except Total Deposition is reported in g/m^2/day.
NA Not Applicable.

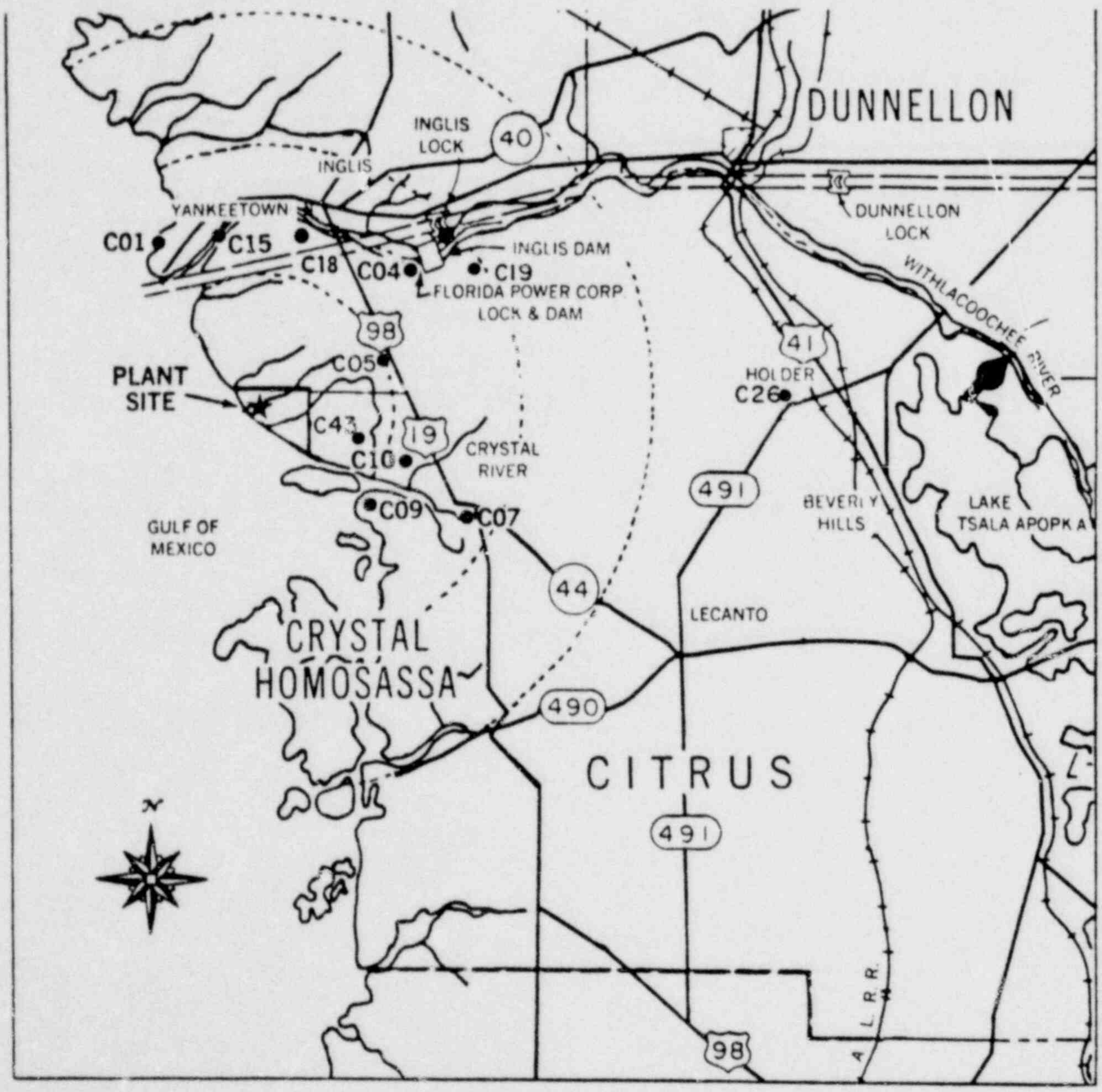
Table 3.2-[illegible]A

DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS

LLD Analysis	Water [pCi/kg]	Airborne Particulate [pCi/m^3]	Fish, Meat, and Poultry [pCi/kg wet]	Milk [pCi/l]	Vegetation [pCi/kg wet]	Soil (pCi/kg dry]	Total Deposition pCi/m^2/da
Gross Beta (b)	2	0.03	NA	NA	NA	NA	NA
H-3 (b)	1200	NA	NA	NA	NA	NA	NA
K-40 (c)	30	0.02	30	NA	30	30	8
Mn-54	20	0.01	10	10	10	10	5
Co-58	15	NA	15	15	15	NA	NA
Fe-59	30	NA	30	NA	NA	NA	NA
Co-60	20	NA	20	20	20	NA	NA
Zn-65	20	0.01	20	NA	20	20	5
Sr-89	10	0.005	NA	10	10	NA	NA
Sr-90	2	0.001	8	2	8	150	NA
Zr-95	20	0.01	20	20	20	20	5
Ru-106	120	0.06	120	NA	120	120	30
I-131	10	0.005	[illegible]	10	10	10	3
Cs-134	15	0.01	15	15	15	15	NA
Cs-137	20	0.01	20	20	20	20	5
Ba-140	50	0.03	50	50	50	50	NA
Ce-144	NA	0.1	NA	NA	NA	NA	NA
Ra-226	300	0.2	300	NA	300	300	NA
Th-232	20	0.01	50	NA	50	50	5

(a) per University of Florida
(b) Gross Beta and H-3 are reported in pCi/l.
(c) K-40 is reported in g/kg except Total Deposition is reported in g/m^2/day.
NA Not Applicable.

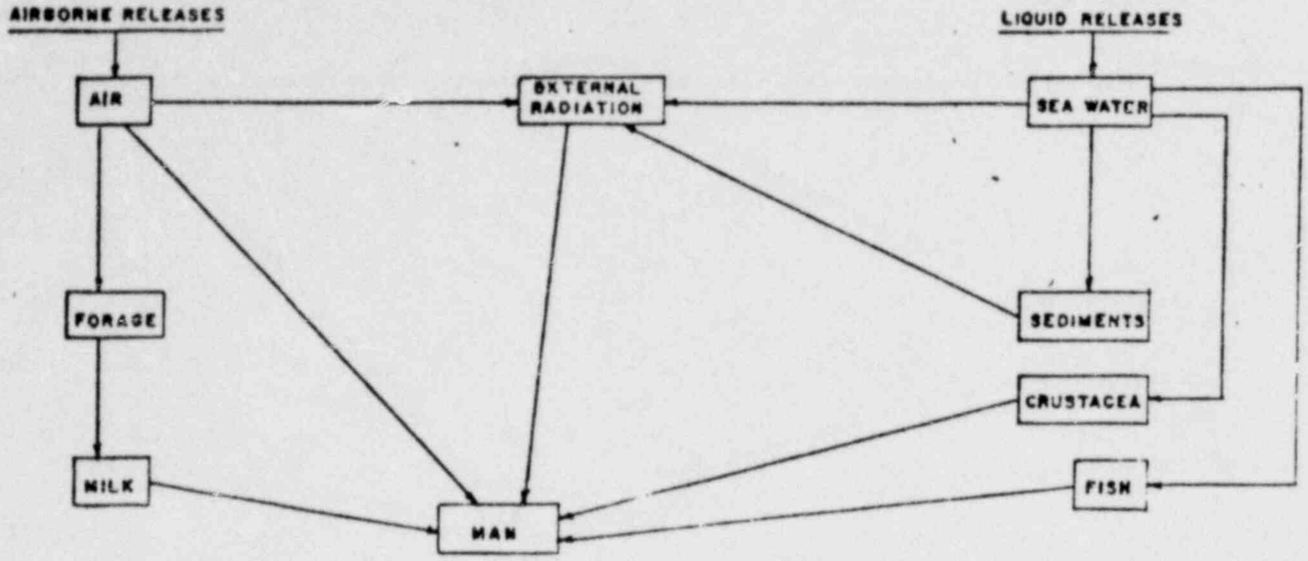
C47 in Gainesville
52 Mi. NNE



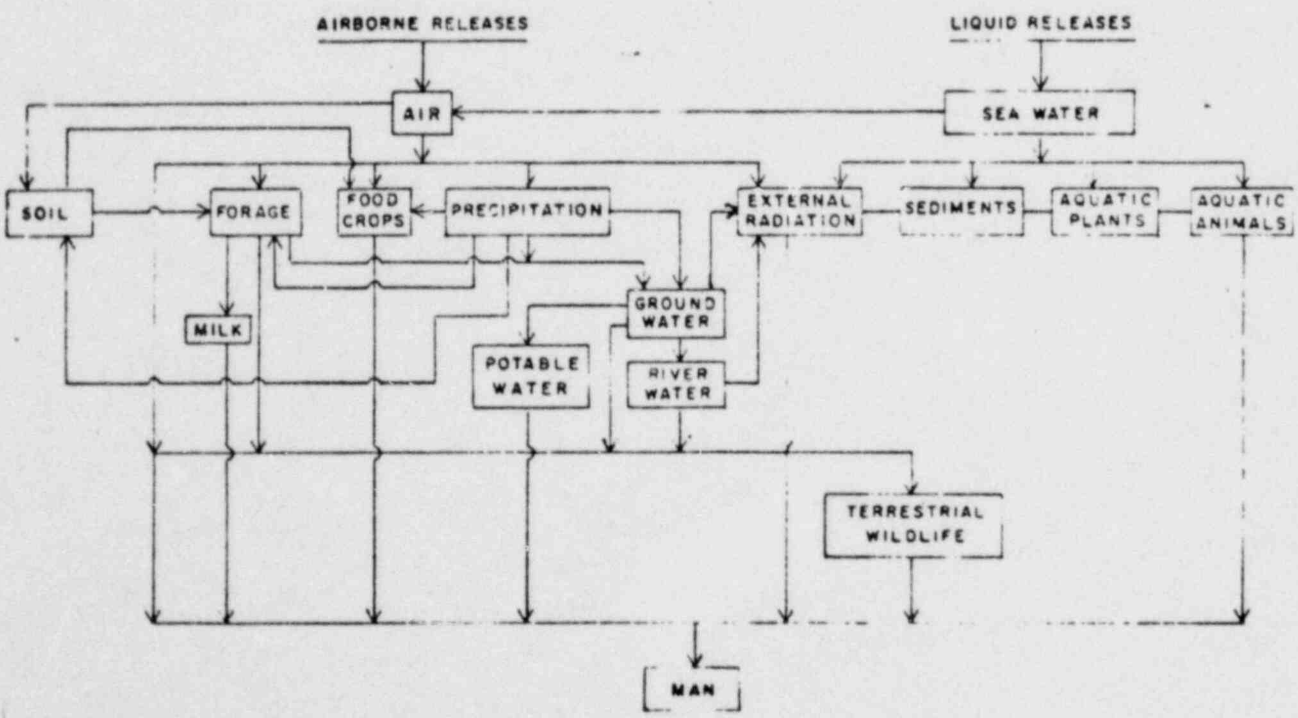
OFF-SITE OPERATIONAL RADIOLOGICA
PROGRAM SAMPLING SITES

CRYSTAL RIVER UNIT 3

FIGURE 3.2-2



CRITICAL PATHWAYS



OTHER MONITORED PATHWAYS

Figure 3.2-1 Environmental Media and Exposure Pathways

Detailed specifications for critical pathway environmental media are divided in three major protection elements: (1) milk and/or green leafy vegetable samples because of the census requirements; (2) external radiation; and (3) all other media.

3.2.1 Milk and Green Leafy Vegetables Census

Specification

3.2.1.1 A census of animals producing milk for human consumption shall be conducted semiannually to determine their location and number with respect to the site. The census shall be conducted under the following conditions:

a. Within a 1-mile radius from the plant site enumeration by a door-to-door or equivalent counting technique.

b. Within a 5-mile radius for cows and a 5-mile radius for goats, enumeration by using referenced information from county agricultural agents or other reliable sources.

If it is learned from this census that animals are present at a location which yields a calculated thyroid dose greater than from previously sampled animals, the new animals shall replace the animals previously sampled.

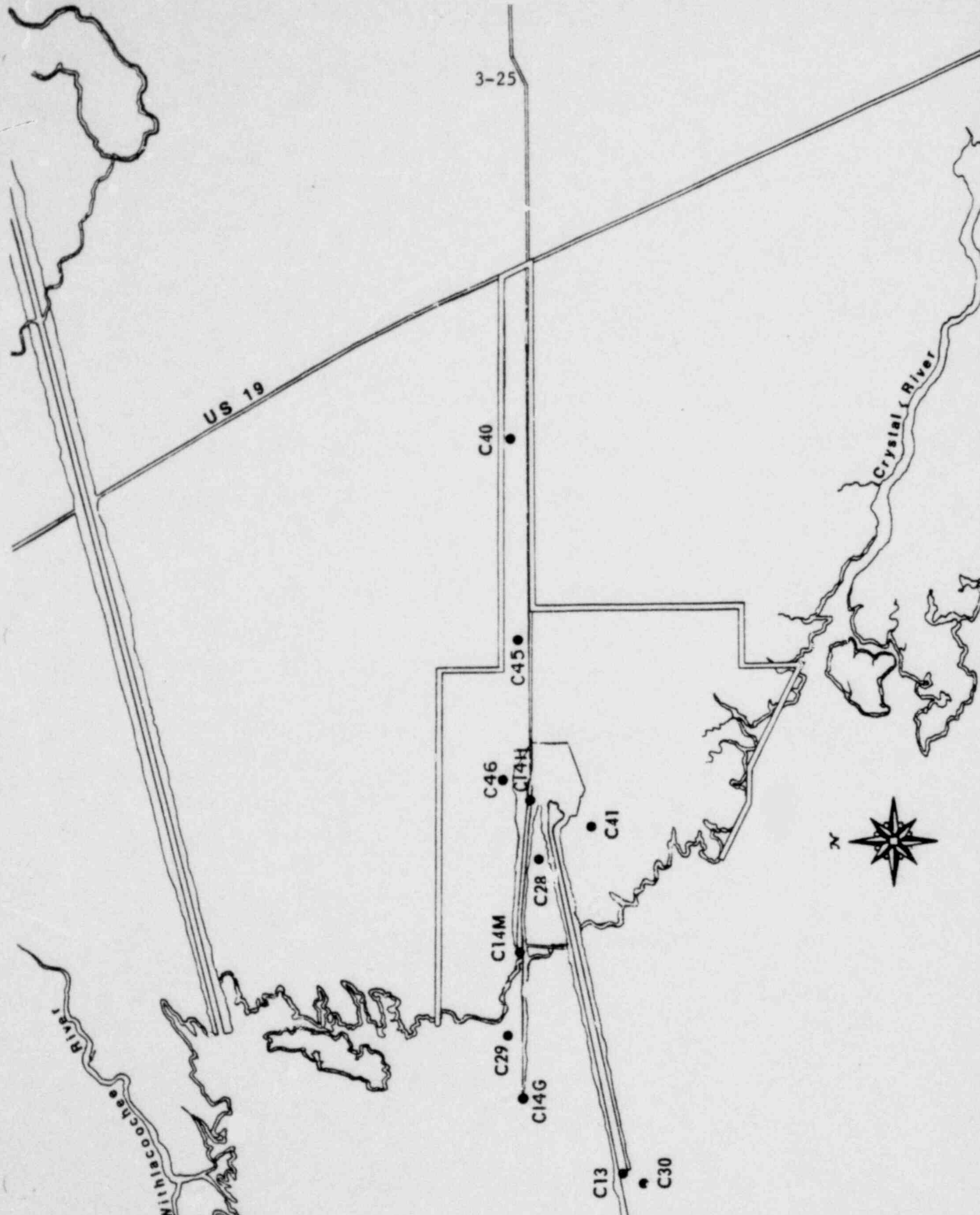
Also, any location from which milk can no longer be obtained may be dropped from the surveillance program after notifying the NRC in writing that milk-producing animals are no longer present at that location.

3.2.1.2 A census of gardens producing fresh leafy vegetables for human consumption (e.g., lettuce, spinach, etc.) shall be conducted annually to determine their location with respect to the site. This census is limited to gardens having an area of 500 square feet or more and shall be conducted under the following conditions:

a. Within a 1-mile radius of the plant site, enumeration by a door-to-door or equivalent counting technique.

b. Within a 5-mile radius of the plant site, enumeration by using referenced information from county agricultural agents or other reliable sources if no milk producing animals are located in the vicinity of the site as determined in 3.2.1.1.

If this census indicates the existence of a garden at a location yielding a calculated thyroid dose greater than that from the previously sampled garden, the new garden shall replace the garden previously sampled. Also,



ON-SITE OPERATIONAL RADIOLOGICA
PROGRAM SAMPLING SITES

CRYSTAL RIVER UNIT 3

FIGURE 3.2-3

If individual milk samples show I-131 concentrations of 10 picocuries per liter or greater, a plan shall be submitted within one week advising of the proposed action to ensure the plant-related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

If green leafy vegetable samples collected in lieu of milk over a calendar quarter show average concentrations of I-131, 220 picocuries per kilogram or greater, a plan shall be submitted within thirty (30) days advising of the proposed action to ensure the plant-related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

If individual green leafy vegetable samples collected in lieu of milk show I-131 concentration of 440 picocuries per kilogram or greater, a plan shall be submitted within one week advising of the proposed action to ensure the plant-related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

External Radiation

The results will be reported in routine reports as specified in Section 5.6.1.

any location from which fresh leafy vegetables can no longer be obtained may be dropped from the surveillance program after notifying the NRC in writing that such vegetables are no longer grown at that location.

3.2.2 Media Other Than External Radiation

Specification

Samples shall be taken from locations and at frequencies listed in Table 3.2-2 and will be analyzed according to the routine listed in Table 3.2-2 using procedures which shall provide concentration values with LLD which are equal to or less than those listed in Table 3.2-5.

The preoperational environmental surveillance results (Table 3.2-1) have been summarized. The control station value for each media, radioisotope and station will be defined as either (1) the upper 95% percentile value from the preoperational program or (2) the upper 95 percentile lue from operational stations outside of the plant's influence, whichever is smaller.

3.2.3 External Radiation

Specification

Ambient external radiation levels shall be measured at locations and frequencies listed in Table 3.2-2 using procedures which will provide radiation level values with LLD increases over preoperational mean background which are equal to or less than that listed in Table 3.2-5.

Reporting Requirements

The results of the radiological monitoring program shall be reported on a routine basis as specified in Section 5.6.1. In addition, measurements of radioactivity in critical pathway environmental media samples shall be reported on a nonroutine basis as described in Section 5.6.2.

3.2.4 Milk and Green Leafy Vegetables

If milk samples collected over a calendar quarter show average concentrations of 4.8 picocuries per liter or greater, a plan shall be submitted within thirty (30) days advising of the proposed action to ensure the plant-related annual doses will be within the design objective of 15 mrem/yr to the thyroid of any individual.

4.2 INTAKE VELOCITY DETERMINATION

Objective

To measure the velocity of water at the intake screens and to verify the validity of the calculated value of intake velocity.

General Approach and Schedule

Velocity currents shall be measured to the nearest 0.1 ft/sec using a 711 mash-McBirney electromagnetic induction currentmeter or its equivalent. Measurements shall be made at the surface, 3' and 6' elevations at each side of the canal, and 3' intervals from the surface at mid-canal during normal operation of all three units within a period of one year after startup of Unit 3.

4.3 STUDY OF EROSION IN THE DISCHARGE SYSTEM

Objective

To study scouring and deposition in the discharge system.

General Approach and Schedule

It is expected that doubling of the discharge volume flow rate by the addition of Unit 3 will nearly double discharge canal velocities and thus could increase the scouring already occurring in the canal. If the canal turbidities were doubled, they would be approximately the same as those of the Withlacoochee River Barge Canal Waters. Much of the scoured canal sediment would be deposited about a mile and a half down the canal and to the west where the currents diverge. Since the scoured discharge canal particles are expected to be larger in grain size than the sediments suspended in the slower Withlacoochee waters, near-canal regions would be affected by scoured canal suspended sediments. These particles would most likely be redeposited once the canal currents slowed upon divergence. The contribution to the turbidity characteristics of the characteristics of the region will continue to be dominated by the suspended loads associated with the Withlacoochee River-Barge Canal complex.

Sediment levels and particle size are being investigated in areas of concern and changes in the community structure can be detected.

4.0 SPECIAL SURVEILLANCE, RESEARCH, OR STUDY ACTIVITIES

4.1 THERMAL PLUME DETERMINATION DURING UNIT 3 OPERATION

Objective

To establish the location and size of the thermal plume during normal operation, under conditions of high and low tide and maximum and minimum intake temperature; to provide data to verify the mathematical and physical models so that good predictions of isotherm location under all conditions will be possible and to establish the operational monitoring system: to verify previous calculations which predict the size and location of the effluent thermal plume.

General Approach and Schedule

Intensive field surveys shall be conducted twice during the first year of operation. Specifically, the surveys will be done during the months of July or August when the maximum intake temperature is observed and during the months of December or January for contrast when the minimum intake temperature is observed. The thermal field measurements shall be made in sufficient locations to cover the full extent of the thermal plume.

Salinity measurement may be required in order to effectively decouple the plume from ambient isotherms. During the tests the behavior of the plume during both phases of the tidal cycle shall be tested. The measurements should allow for construction of the isothermal maps with 1.0°F above ambient contour intervals. These tests shall be carried out with all three units operational and under at least 80 percent of full capacity. During the surveys the following conditions shall be recorded as needed to assess the extent of the thermal plume and its correspondence to a computer run with parallel parameters: (a) plant conditions (condenser flows, intake temperature, discharge temperature, loading, etc.) of all three units, (b) hydrological conditions (tidal stage, salinity traverses, etc.), (c) meteorological conditions (wet and dry bulb temperature, humidity, windspeed, wind direction, solar radiation, etc.).

The field survey measurements shall be compared to the results of the predicted computer runs. Any modifications needed in either the physical model or the mathematical model will then be incorporated in the models. The models will then be available to use in the evaluation of any abnormal environmental occurrence or other modifications in plant system or equipment performance.

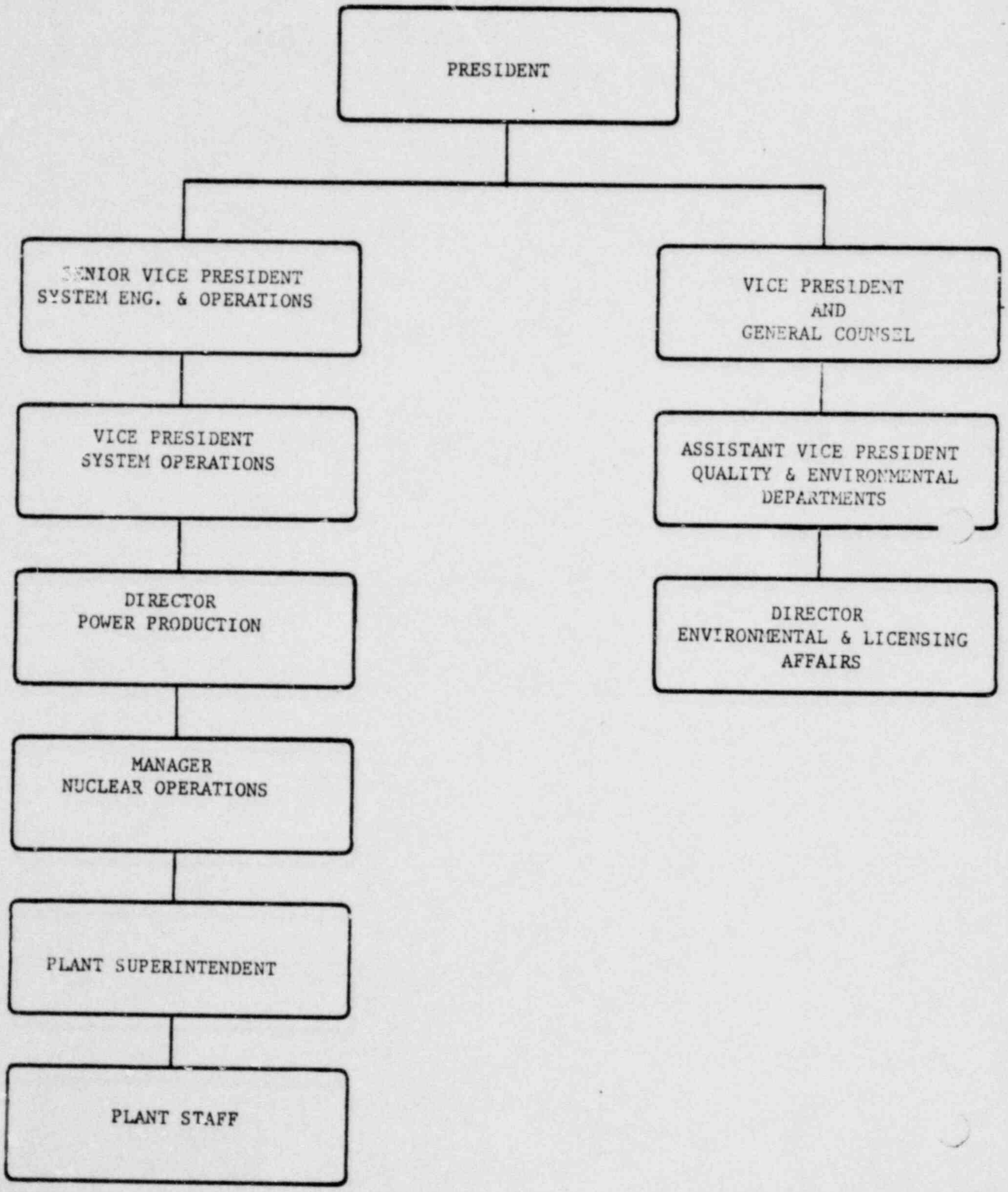


Figure 5.1-1 Organization for implementing Environmental Technical Specifications

5.0 ADMINISTRATIVE CONTROLS

Objective

To define the organization, assign responsibilities, describe the environmental surveillance procedures, provide for a review and audit function, and prescribe the reporting requirements in order to insure continuing protection of the environment and implement the Environmental Technical Specifications.

5.1 ORGANIZATION

The organization responsible for environmental protection, environmental monitoring and the implementation of the Environmental Technical Specifications, both prior to and following the issuance of an operating license for Crystal River Unit 3, is shown on Figure 5.1-1.

5.2 RESPONSIBILITY

The responsibility for the conduct of the preoperational environmental monitoring program described in Section 3 and special studies described in Section 4 is that of the Quality and Environmental Department under the direction of the Director of Environmental and Licensing Affairs. Upon the issuance of an operating license, the responsibility for the conduct of the operational environmental monitoring program and the implementation of Environmental Technical Specifications becomes the responsibility of the System Operations Department.

The plant organization is responsible for the development of Operating and Surveillance Procedures described generally in Section 5.5 and supplying field data to the Manager, Nuclear Operations as required by Sections 2, 3 and 4 of the Environmental Technical Specifications.

The Manager, Nuclear Operations is responsible for consultant contracts, State and local regulatory agreements, assembly of data, preparation and review of reports required by Section 5.6 of these Environmental Technical Specifications, and making recommendations to improve environmental protection practices.

All reports and correspondence with the NRC regarding the Environmental Technical Specifications shall be approved and signed by the Director, Power Production. The Nuclear Plant Superintendent shall, however, make reports by telephone and telegraph of any incident or occurrence requiring reporting within 24 hours or less, as required in Section 5.6.

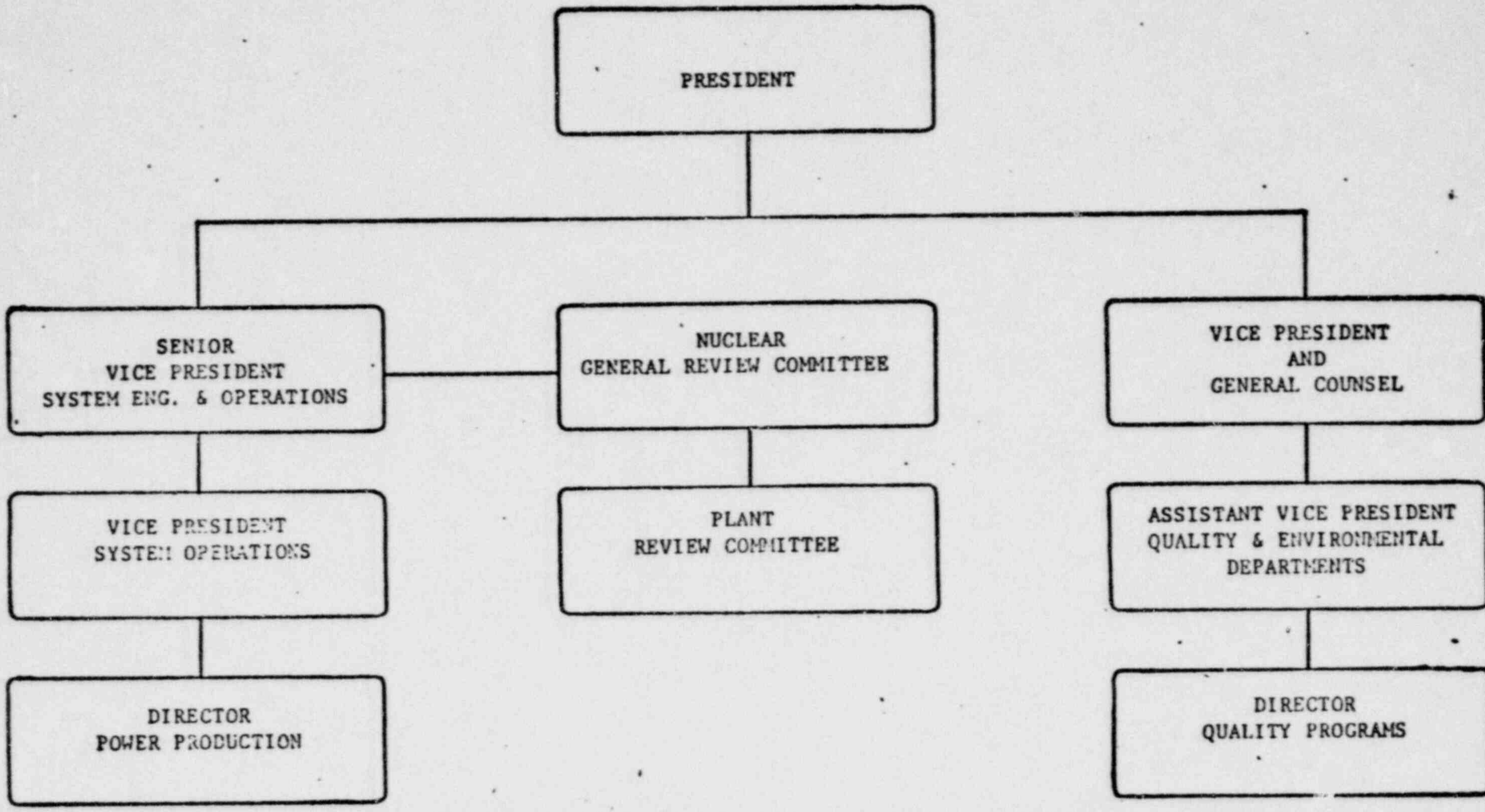


Figure 5.3-1 Organization for Independent Review and Audit

5.3 REVIEW AND AUDIT

The FPC corporate Quality Assurance organization under the direction of the Director of Quality Programs has the responsibility of auditing the implementation of the environmental monitoring and surveillance programs and auditing conformance to procedures and Environmental Technical Specification requirements. Audits shall be conducted at least once per year. The audit shall also include contractor operations. Figure 5.3-1 shows the reporting path for the Director of Quality Programs that is independent of the System Operations Department.

The Nuclear General Review Committee also has a reporting path independent of the System Operations Department as shown on Figure 5.3-1. This Committee will normally fulfill its responsibilities by conducting reviews and reporting the results of these reviews to the Senior Vice President, System Engineering and Operations. However, in accordance with the Nuclear General Review Committee Charter, the Nuclear General Review Committee has the authority to conduct audits of any portion of the Production Department's Quality Control Program.

The Nuclear General Review Committee functions as they relate to the Environmental Technical Specifications are as follows:

a. Review and make recommendations on proposed changes to the Environmental Technical Specifications and the evaluated impact of the changes.

b. Review proposed changes or modifications to plant systems or equipment and the evaluated impact which would require a change in the procedures described in (d) below (or which would affect the evaluation of the plant's environmental impact as described in Section 5.4.2(B)).

c. Review reported instances of violations of Environmental Technical Specifications, the reaching of specified reporting levels, and abnormal environmental occurrences. Where investigation indicates, evaluate and formulate recommendations to prevent recurrence.

subsequently reviewed and approved by the Plant and Nuclear General Review Committees. All procedures and changes to procedures utilized by contractors to implement the environmental monitoring programs described in Section 3 shall be reviewed and approved by the Manager, Nuclear Operations.

5.5.3 Prior to special tests or changes:

a. If the Nuclear Plant Superintendent decides to make a change in the facility or Operating Procedures, or to conduct a test or experiment, and concludes that the proposed change, test, or experiment does not involve a change in the Environmental Technical Specifications or an unreviewed environmental impact question, he may order the change, test, or experiment to be made, shall enter a description thereof in the operating records of the facility, and shall send a copy of the instructions pertinent thereto to the Chairman of the Nuclear General Review Committee for review.

b. If the Nuclear Plant Superintendent desires to make a change in the facility or Operating Procedures, or to conduct a test or experiment which in his opinion might involve a change in the Environmental Technical Specifications, or involve an unreviewed environmental impact question, he shall not order such change, test, or experiment until he has referred the matter to the Nuclear General Review Committee for review and report. If the Committee is of the opinion that the proposed change, test, or experiment does not require approval by the Nuclear Regulatory Commission under the terms of said license, it shall so report in writing to the Nuclear Plant Superintendent, together with a statement of the reasons for the Committee decision and the Nuclear Plant Superintendent may then proceed with the change, test, or experiment. If on the other hand the Committee is of the opinion that approval of the Nuclear Regulatory Commission is required, the Committee shall so report to the Nuclear Plant Superintendant, who will proceed through established procedures to obtain Nuclear Regulatory Commission approval of the proposed change, test or experiment. The Nuclear General Review Committee shall thus withold approval of the proposed activity until supplied with written documentation of Nuclear Regulatory Commission approval.

5.6 PLANT REPORTING REQUIREMENTS

5.6.1. Routine Reports

A. Annual Environmental Operating Report

(1) Nonradiological Volume

A report on the nonradiological environmental surveillance programs for the previous 12 months of operation shall be

d. Review and compare the Standard Technical Specifications and the Environmental Technical Specifications to avoid conflicts and maintain consistency.

5.4 ACTION TO BE TAKEN IF LIMITING CONDITION FOR OPERATION IS EXCEEDED

5.4.1 Immediate remedial actions as permitted by these environmental technical specifications shall be implemented until such time as the limiting condition for operation is met.

5.4.2 The occurrence shall be promptly reported to the Chairman of the Nuclear General Review Committee and investigated as specified in Section 5.3.

5.4.3 The Nuclear General Review Committee shall prepare and submit promptly a separate report for each occurrence in writing to the Senior Vice President, System Operations. The report shall describe the circumstances leading to and resulting from the occurrence, and shall recommend appropriate action to prevent or reduce the probability of repetition

5.4.4 The Director, Power Production, shall report the occurrence to the NRC as specified in Section 5.6.2.

5.5 PROCEDURES

5.5.1 Explicit written procedures, including applicable check-off lists and instructions, shall be prepared for the implementation of the monitoring requirements described in Sections 2 and 3, approved as specified in Section 5.5.2, and adhered to for operation of all systems and components involved in carrying out the effluent release and environmental monitoring programs. Procedures shall include sampling, instrument calibration, analysis, and action to be taken when limits are approached or exceeded. Calibration frequencies and standards for instruments used in performing the measurements shall be included. Testing frequency of alarms shall be included. These frequencies shall be determined from experience with similar instruments in similar environments and from manufacturers' technical manuals.

5.5.2 All procedures implemented by plant staff personnel described in Section 5.5.1 above, and changes thereto, shall be reviewed as specified in Section 5.3 and approved by the Nuclear Plant Superintendent prior to implementation. Temporary changes to procedures which do not change the intent of the original procedure may be made, provided such changes are approved by two members of the plant management staff, one of whom holds a senior operator's license. Such changes shall be documented, Results of all radiological environmental samples taken shall be summarized on an annual basis following the format of Table 5.6-1. 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.

B. Semiannual Operating Report - Radioactive Effluents

A report on the radioactive discharges released from the site during the previous 6 months of operation shall be submitted to the Director of the Office of Inspection and Enforcement (with a copy to Director, Office of Nuclear Reactor Regulation) as part of the Semiannual Operating Report within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant as outlined in USNRC Regulatory Guide 1.21, with data summarized on a quarterly basis following the format of Appendix B thereof.

The report shall include a summary of the meteorological conditions concurrent with the release of gaseous effluents during each quarter as outlined in USNRC Regulatory Guide 1.21, with data summarized on a quarterly basis following the format of Appendix B thereof. Calculated offsite dose to humans resulting from the release of effluents and their subsequent dispersion in the atmosphere (Regulatory Guide 1.109) shall be reported in accordance with Regulatory Guide 1.21.

5.6.2 Non-Routine Reports

A. Limiting Condition for Operation Exceeded

In the event that a limiting condition for operation is exceeded including any unplanned release of radioactive material from the site, or an event involving a significant adverse environmental impact occurs, a report will be made within 24 hours by telephone and telegraph to the Director of the Office of Inspection and Enforcement followed by a written report with a copy to the Director, Office of Nuclear Reactor Regulation within 15 days. The telegraph report will quantify the occurrence, its causes and, if aspects of the Crystal River Unit 3 operation are among the causes, planned remedial action to the extent possible. The written report will fully describe the occurrence and will describe its causes and corrective action as fully as possible.

submitted to the Director of Inspection and Enforcement (with copy to Director of Nuclear Reactor Regulation) as a separate Volume (#1) of the Annual Environmental Operating Report within 90 days after January 1 of each year. The period of the first report shall begin with the date of initial criticality. The report shall include summaries, interpretations, and statistical evaluation of the results of the nonradiological environmental surveillance activities (Section 3.0) and the environmental monitoring programs required by limiting conditions for operation (Section 2.0) for the report period. 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 shall be provided. 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. Special surveillance, research, or study activities reports shall be submitted to the Director of Inspection and Enforcement (with copy to Director, Office of Nuclear Reactor Regulation) within 15 months of the commercial operation of the plant.

(2) Radiological Volume

A report on the radiological environmental surveillance programs for the previous 12 months of operation shall be submitted to the Director of Inspection and Enforcement (with copy to Director, Office of Nuclear Reactor Regulation) as a separate volume (#2) of the Annual Environmental Operating Report within 90 days after January 1 of each year. The period of the first report shall begin with the date of initial criticality. The report 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. 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.

B. Nonradiological Report Levels

In the event that a nonradioactive reporting level is reached, a written report shall be made within 30 days to the Director of the Office of Inspection and Enforcement with a copy to the Director of the Office of Nuclear Reactor Regulation. The report shall describe, analyze and evaluate the occurrence, including extent and magnitude of impact; describe the cause of the occurrence; and if aspects of the Crystal River Unit #3 operation are among the causes, indicate the correction action taken to preclude repetition of the occurrence.

C. Radiological Reporting Levels

In the event a report level specified below is reached, a report shall be made within the designated time period to the Director of Inspection and Enforcement with a copy to the Director of Office of Nuclear Reactor Regulation.

(1) Radioactive Discharge

If measured rates of release of radioactivity in the environment, averaged over a calendar quarter, exceed the design objective rates as specified in specifications 2.4.1.H for liquid effluents and in 2.4.2.C for airborne effluents, a report of the causes of the release rates and of a proposed program of action to reduce the release rates will be submitted within 30 days from the end of the quarter during which the release occurred.

(2) Radiological Environmental Monitoring

If a single measured value of radioactivity concentrations in critical pathway environmental medium samples identified in Section 3.2 exceeds ten times the control station value as defined in Section 3.2, a written notification including an evaluation of any release conditions, environmental factors, or other aspects necessary to explain the anomalous result 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.*

*Confirmation is defined in Regulatory Guide 4.8.

Example Data Presentation[d]

TABLE 5.1

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM SUMMARY

Name of Facility ______________________ Docket No. ______________

Location of Facility ______________________ Reporting Period ______________
(County, State)

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: Name, Distance and Direction	Location with Highest Annual Mean: Mean (f)[b] Range[b]	Control Locations Mean (f)[b] Range[b]	Number of Nonroutine Reported Measurements[c]
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) (0.03-0.13)	<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. 8/73), pp. D-08-01, 02, 03.
[b]Mean and range based upon detectable measurements only. Fraction of detectable measurements at specified locations is indicated in parentheses. (f)
[c]Nonroutine reported measurements are defined in Section 5.6.2.h.
[d]Note: The example data are provided for illustrative purposes only.

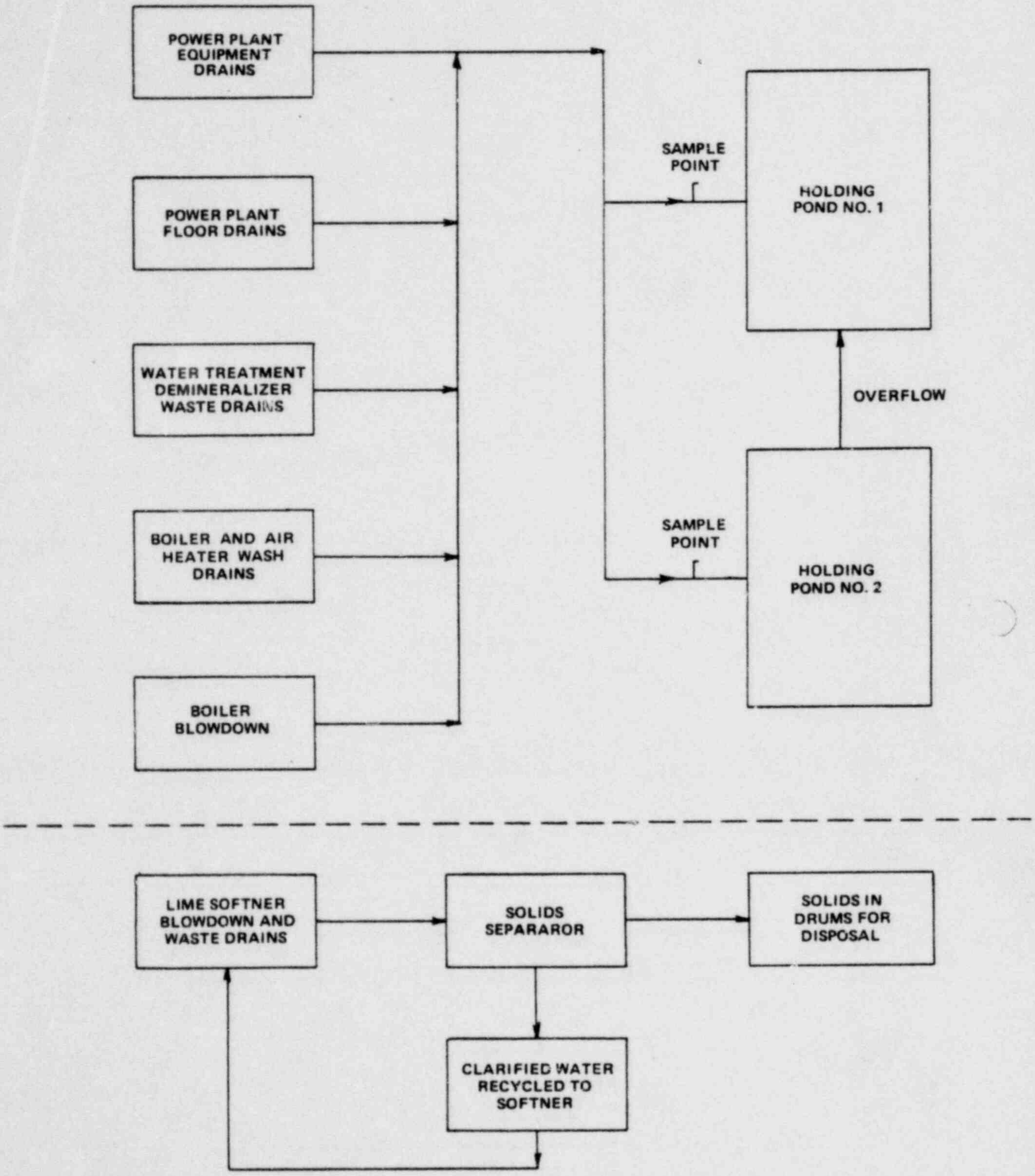


Figure 5.8-1 Chemical-Industrial Waste Water Treatment System