UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The applications for amendment by Florida Power Corporation, et al (the licensees) dated July 15, October 11, and November 8, 1977, and February 17, 1978, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the applications, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 17, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: January 4, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendices "A" and "B" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Appendix "A" Pages	Appendix "B" Pages
2-1	1-1
2-5	2-1
2-7	2-2
B 2-2	2-12
B 2-5	2-22
3/4 4-1	
3/4 4-2	
3/4 5-4	
3/4 5-5	
3/4 6-12	
3/4 6-13	
B 3/4 4-1	
B 3/4 5-2	
B 3/4 5-3 (added)	
B 3/4 6-3	

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 Figur 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 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.

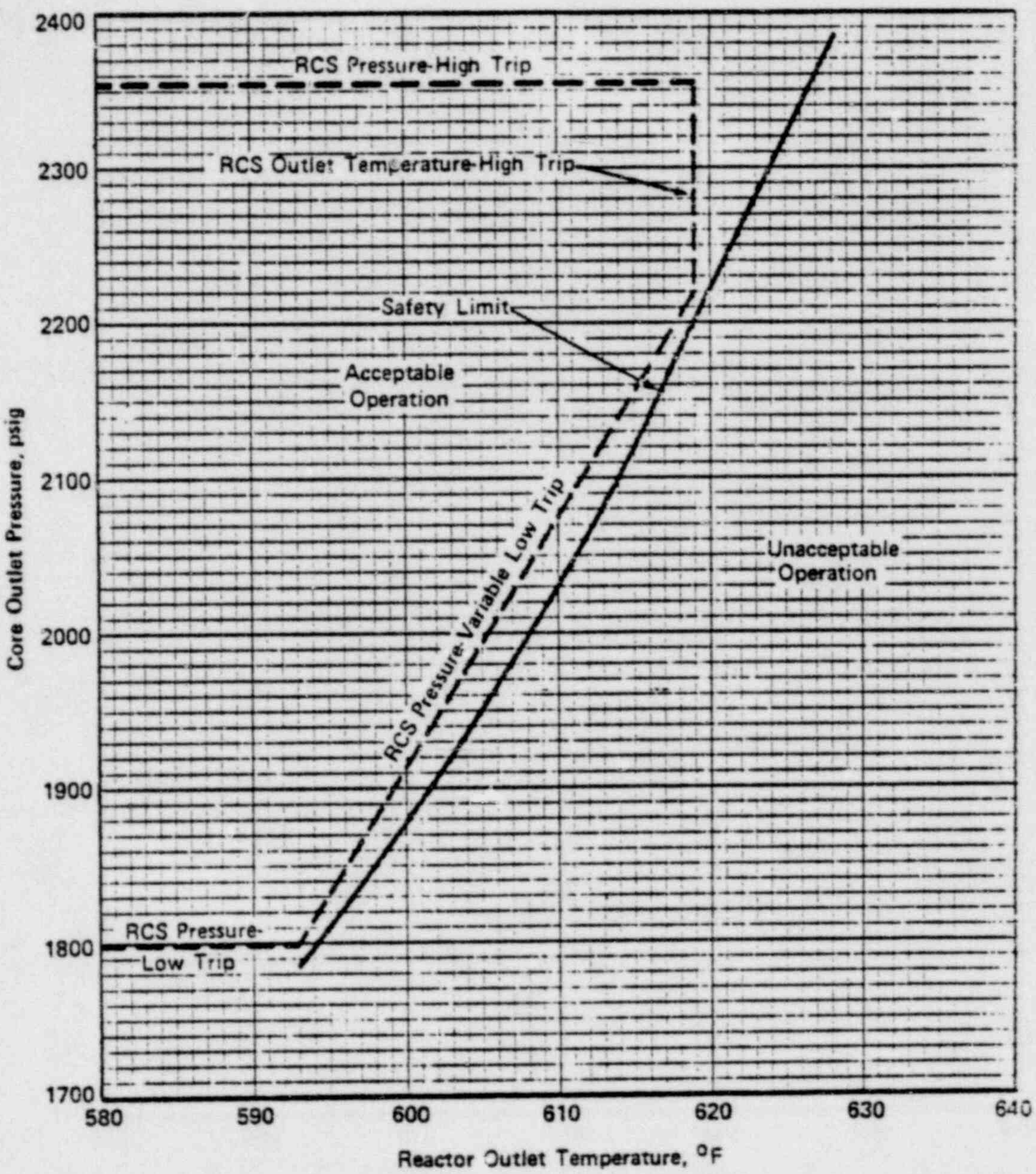

Figure 2.1-1
Reactor Core Safety Limit

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 $\leq$ 78% of RATED THERMAL POWER with three pumps operating	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating $\leq$ 78% of RATED THERMAL POWER with three pumps operating
3. RCS Outlet Temperature-High	$\leq$ 619°F	$\leq$ 619°F
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE (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.

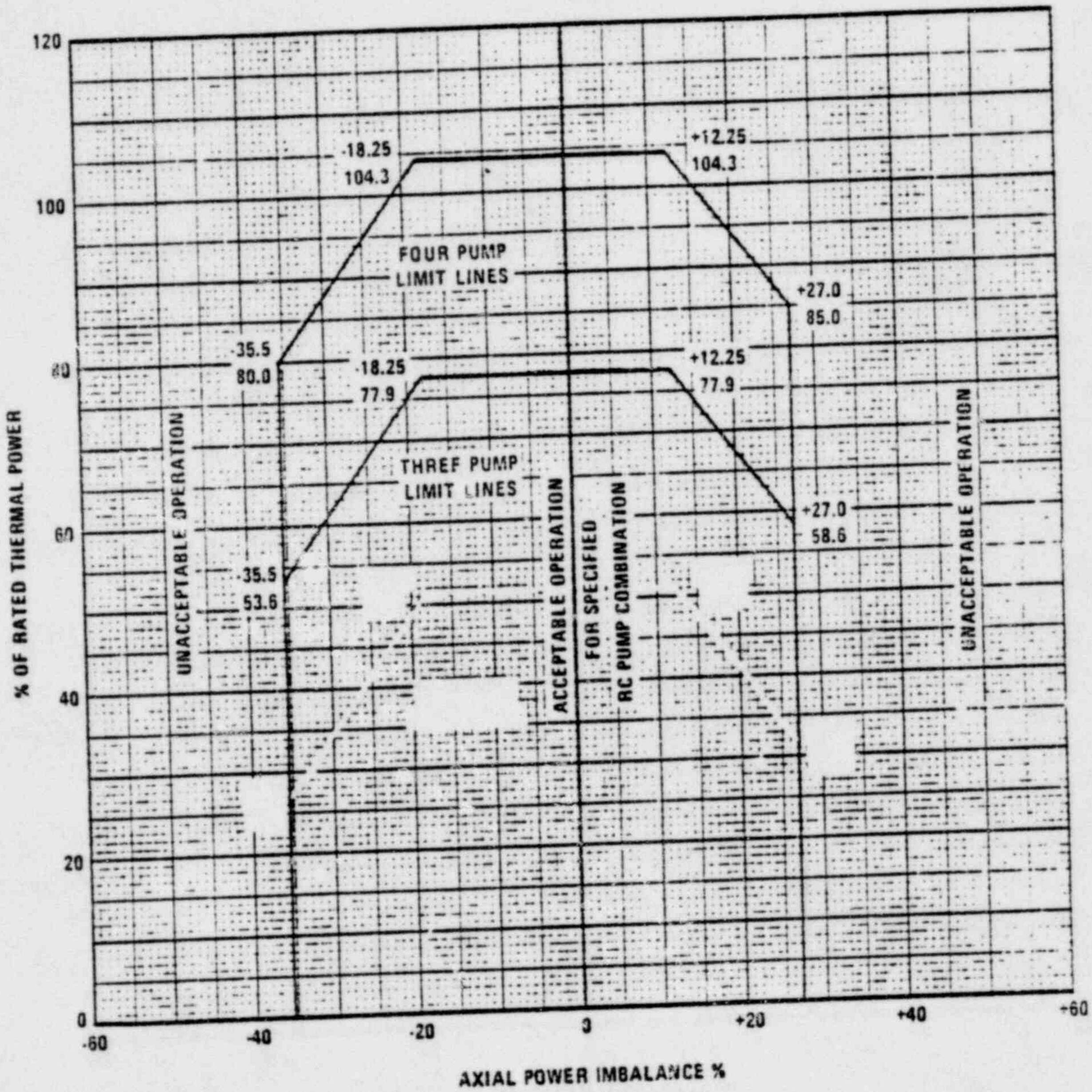

Figure 2.2-1

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

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 BAW-2 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 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The 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 137.89 x 10^6 lbs/hr, which is 105% of 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:

$$F_Q^N = 2.57; \quad F_{\Delta H}^N = 1.71; \quad F_Z^N = 1.50$$

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.

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 $F_Q^N = 2.57$ 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 and 2 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps and three pumps, 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 22%, whichever condition is more restrictive.

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

The DNBR as calculated by the BAW-2 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 22% is justified on the basis of experimental data.

LIMITING SAFETY SYSTEM SETTINGS

BASES

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip $\leq$ 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 accommodate 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 $\geq$ 104.3% and reactor flow rate is 100%, or flow rate is $\leq$ 95.9% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is $\geq$ 77.9% and reactor flow rate is 74.7%, or flow rate is $\leq$ 71.9% and power is 75%.

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

BASES

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 1.043% 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.

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

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.

*See Special Test Exception 3.10.3.

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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. By verifying the correct position of each mechanical position stop for the following HPI stop check valves prior to restoring the HPI system to OPERABLE status following periodic valve stroking or maintenance on the valves.

   1. MUV-2,
   2. MUV-6,
   3. MUV-10

d. By verifying that the flow switches for the following LPI throttle valves operate properly prior to restoring the LPI system to OPERABLE status following periodic valve stroking or maintenance on the valves.

   1. DHV-110,
   2. DHV-111

e. 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. Verifying the correct position of each mechanical position stop for each of the stop check valves listed in Specification 4.5.2.c.

3. Verifying that the flow switches for the throttle valves listed in Specification 4.5.2.d operate properly.

4. 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.

5. Verifying a total leak rate $\leq$ 6 gallons per hour for the LPI system at:

   a) Normal operating pressure or a hydrostatic test pressure of $\geq$ 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.

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

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

g. Following completion of HPI or LPI system modifications that could have altered system flow characteristics, by performance of a flow balance test during shutdown to confirm the following injection flow rates:

HPI System - Single Pump	LPI System - Single Pump
Injection Leg $A_1 \geq$ 250gpm @600psig	Injection Leg A-2800 to 3100 gpm
Injection Leg $A_2 \geq$ 250gpm @600psig	
Injection Leg $B_1 \geq$ 250gpm @600psig	Injection Leg B-2800 to 3100 gpm
Injection Leg $B_2 \geq$ 250gpm @600psig	

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.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

   2. ≥ 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 the spray additive tank containing at least 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).

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 tank, and

2. Verifying the concentration of the NaOH solution by chemical analysis.

CONTAINMENT SYSTEMS

SPRAY ADDITIVE SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 The spray additive system shall be OPERABLE with the spray additive tank containing at least 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).

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 tank, and

   2. Verifying the concentration of the NaOH solution by chemical analysis.

SURVEILLANCE REQUIREMENTS (Conti[illegible]ed)

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

CONTAINMENT SYSTEMS

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.

3/4.4 REACTOR COOLANT SYSTEM

BASES

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 loop, 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 22%, 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

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.

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.

The purpose of these surveillance requirements is to provide assurance that proper ECCS flows will be maintained in the event of a LOCA. Maintenance of proper flow resistance and pressure drop in the piping system to each injection point is necessary to: (1) prevent total pump flow from exceeding runout conditions when the system is in its minimum resistance configuration, (2) provide the proper flow split between injection points in accordance with the assumptions used in the ECCS-LOCA analyses, and (3) provide an acceptable level of total ECCS flow to all injection points equal to or above that assumed in the ECCS-LOCA analyses.

3/4.5.4 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.

BORATED WATER STORAGE TANK (Continued)

The contained water volume limit includes an allowance for water not usable because of tank discharge line location or other physical characteristics. The limits on contained water volume, and boron concentration 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.

3/4.6.2 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 is added to the containment spray in the event of a LOCA. The limits on contained sodium hydroxide 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.

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.

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.

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.

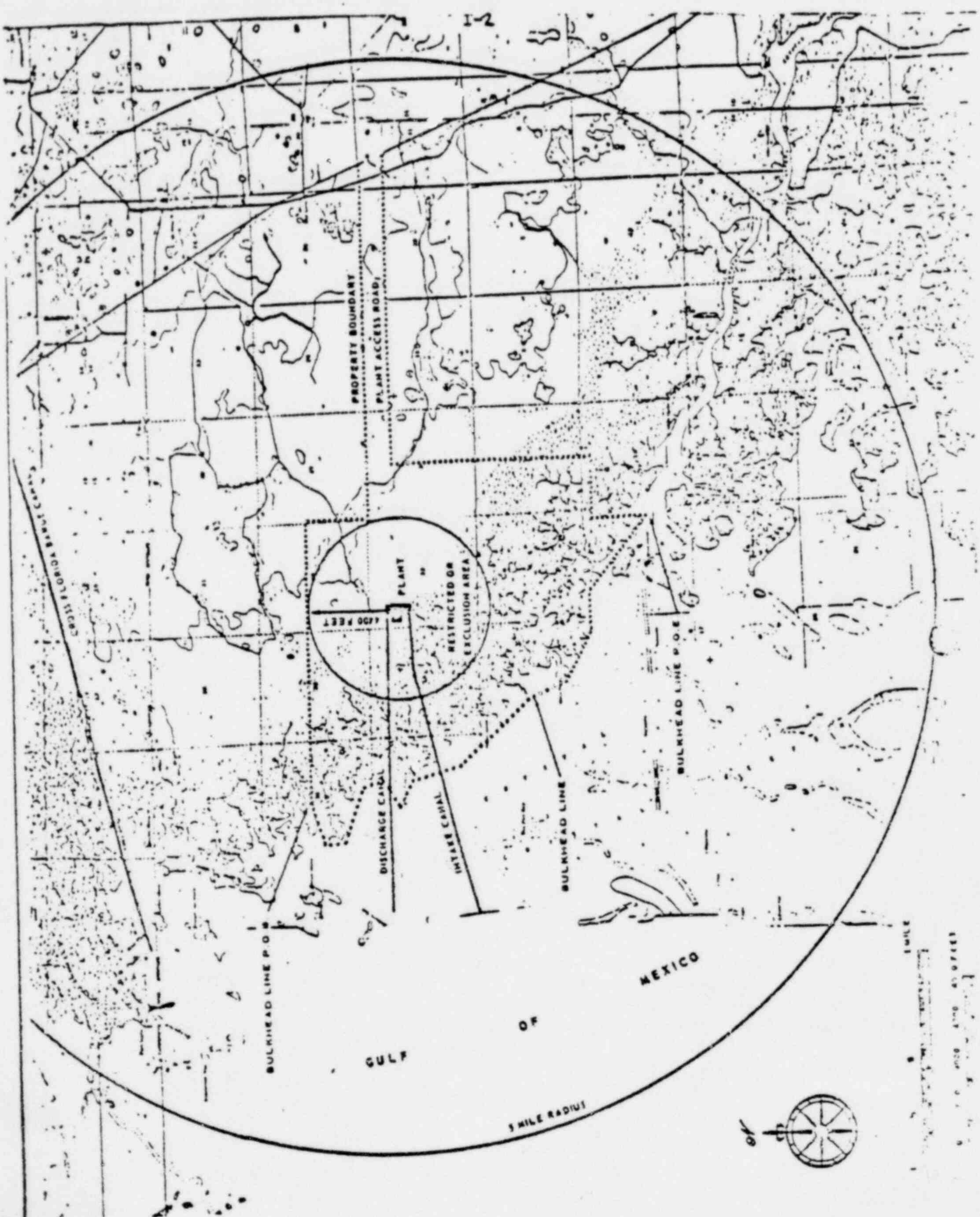

Figure 1.1-1 Point of Discharge

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 Unit 3 during normal operation at all power levels.

Specification

The temperature rise across the unit 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 unit temperature rise shall be monitored by detectors (RTD's 0-200 ± 1°F) located in the inlet and outlet of Unit 3. 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 unit ΔT shall be determined every 2 hours ± 1 hour utilizing local temperature indicators (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 the extreme climatic conditions, especially during tropical storms, sea grass is uprooted from the Gulf of Mexico, requiring temporary shutdown of a circulator 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 unit. 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 inlet and outlet of Unit 3 will provide reliable values of the ΔT across the unit.

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 the 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) The average release rate of noble gases from the site during any 12 consecutive months shall be

$$25\left[Q_{Tv}\overline{N}_v\right] \leq 1$$

and

$$13\left[Q_{Tv}\overline{M}_v\right] \leq 1$$

(3) The average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter shall be such that

$$13\left[3.5 \times 10^4\, Q_v\right] \leq 1$$

(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 rat--, define and initiate a program of action to reduce the rele...e rates to design objective levels listed in Section 2.4, and report these actions to the NRC within 30 days from the end of the quarter during which the releases occurred.

(1) If the average release rate of noble gases from the site during any calendar quarter is such that

$$50\left[Q_{Tv}\overline{N}_v\right] > 1$$

or

$$25\left[Q_{Tv}\overline{M}_v\right] > 1$$

(2) If the average release rate per site of all radioiodines and radioactive materials in particulate form with half-lives greater than eight days during any calendar quarter is such that

$$50 \, [3.5 \times 10^4 Q_v] \quad >1$$

(3) If the amount of iodine-131 rleased 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 measured and recorded per Table 2.4-4. 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

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	Dissolved 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 2.4-4

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	Flow Rate Recorder	Continuous Monitor	Grab Sample Station	Measurement Capabilities NG	I	Part	H-3	Alpha
Process Stream										
Waste Gas Decay Tank	x	x	Continuous	RMA-11	x	x	x	x	x	x
Condenser Vacuum Pump Exhaust	x		Once Per Shift	RMA-12	x	x	x	x	x	x
Building Ventilation Systems										
Reactor Building Purge Exhaust Duct [whenever there is flow]	x	x	Continuous	RMA-1	x	x	x	x	x	x
Auxiliary Building and Fuel Handling Building Exhaust Duct*	x	x	Continuous	RMA-2	x	x	x	x	x	x

*This exhaust includes the radwaste area.