

DN 3016

JULY 12 1983

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Docket No. 50-302

Mr. Walter S. Wilgus
 Vice President, Nuclear Operations
 Florida Power Corporation
 ATTN: Manager, Nuclear Licensing
 & Fuel Management
 Post Office Box 114042, M.A.C H-2
 St. Petersburg, Florida 33733

Dear Mr. Wilgus:

SUBJECT: CYCLE 5 FUEL RELOAD

Crystal River Unit No. 3 Nuclear Generating Plant

The Commission has issued the enclosed Amendment No. 64 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The amendment consists of changes to the Technical Specifications in response to your application dated March 31, 1983, as supplemented on June 17, 1983, June 22, 1983, and July 6, 1983. The amendment revises the Technical Specifications to authorize Cycle 5 operation of CR-3 following refueling. A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's Monthly section of the Federal Register to be published on or about July 20, 1983.

As discussed in our Safety Evaluation, and reported to you by letter dated June 20, 1983, we have evaluated and denied your request to extend all 18-month surveillance requirements to 24 months to correspond to the extended core lifetime for Cycle 5. The staff considers that the present intervals are necessary to ensure confidence in proper operation of the components affected and that flexibility normally exists in accomplishing these surveillances without necessarily having to shutdown for the sole purpose of their accomplishment.

Various other changes to the CR-3 Technical Specifications have been herewith included in response to your request and in conjunction with plant physical improvements. The requested changes relative to plant modifications required by NUREG-0737 are under staff evaluation and will be issued by separate amendment.

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OFFICE
SURNAME
DATE

Mr. Walter S. Wilgus

- 2 -

In response to your June 17, 1983 letter, the staff has reevaluated and revised test requirements for the Reactor Coolant Pump Power Monitor trip. This is discussed in the enclosed Safety Evaluation and reflected in the Technical Specifications.

Sincerely,

**"ORIGINAL SIGNED BY
JOHN F. STOLZ"**

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Enclosures:

- 1. Amendment No. 64
- 2. Safety Evaluation

cc w/enclosures:

See next page

OFFICE	DL: ORB #4	DL: ORB #4	OELD	DL: ORB #4	DL: AD/OR	DL: DIR
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DATE	7/11/83	7/11/83	7/11/83	7/11/83	7/11/83	7/11/83

Crystal River Unit No. 3
Florida Power Corporation

50-302

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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
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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. 64
License No. DPR-72

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Florida Power Corporation, et al. (the licensees) dated March 31, 1983, as supplemented June 17, 1983, June 22, 1983, and July 6, 1983, complies 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 application, 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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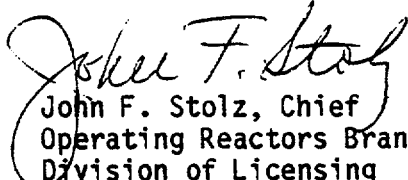
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-72 is hereby amended to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 64, 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


John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 12, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 64

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" 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.

Pages

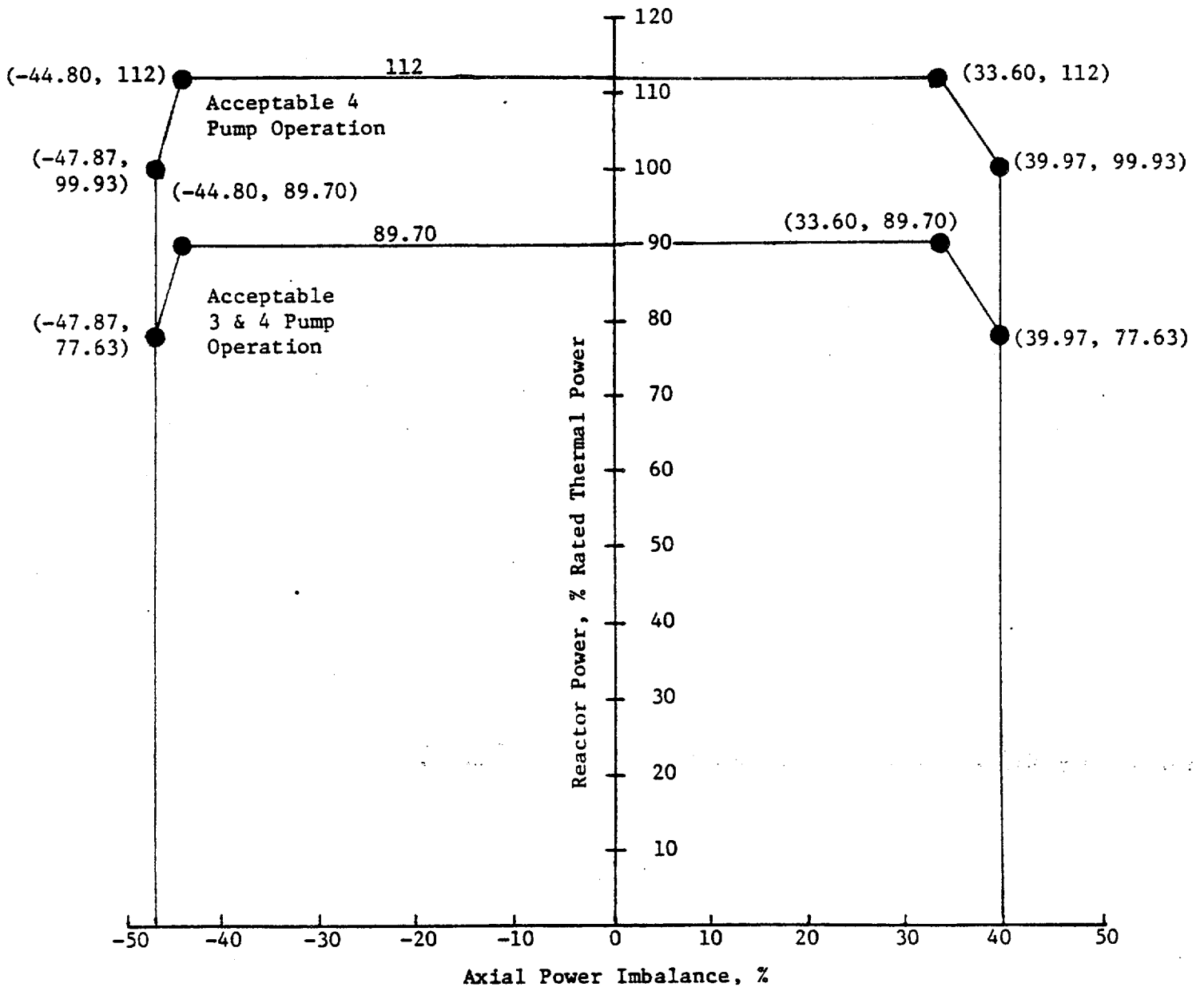
2-3
2-5
2-6
2-7
B2-4
B2-5
B2-6
B2-7
3/4 1-1
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3/4 1-7
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B 3/4 1-2
B 3/4 4-2
B 3/4 7-2

FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT



SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETTINGS

2.2.1 The Reactor Protection System instrumentation settings 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 setting 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 OPERATE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTION UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	$\leq 104.9\%$ of RATED THERMAL POWER with four pumps operating	$\leq 104.9\%$ of RATED THERMAL POWER with four pumps operating
	$\leq 79.92\%$ of RATED THERMAL POWER with three pumps operating	$\leq 79.92\%$ of RATED THERMAL POWER with three pumps operating
3. RCS Outlet Temperature - High	$\leq 618^{\circ}\text{F}$	$\leq 618^{\circ}\text{F}$
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)	≥ 1800 psig	≥ 1800 psig
6. RCS Pressure - High	≤ 2300 psig	≤ 2300 psig
7. RCS Pressure - Variable Low (1)	$\geq (11.59 T_{\text{out } ^{\circ}\text{F}} - 5037.8)$ psig	$\geq (11.59 T_{\text{out } ^{\circ}\text{F}} - 5037.8)$ psig

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

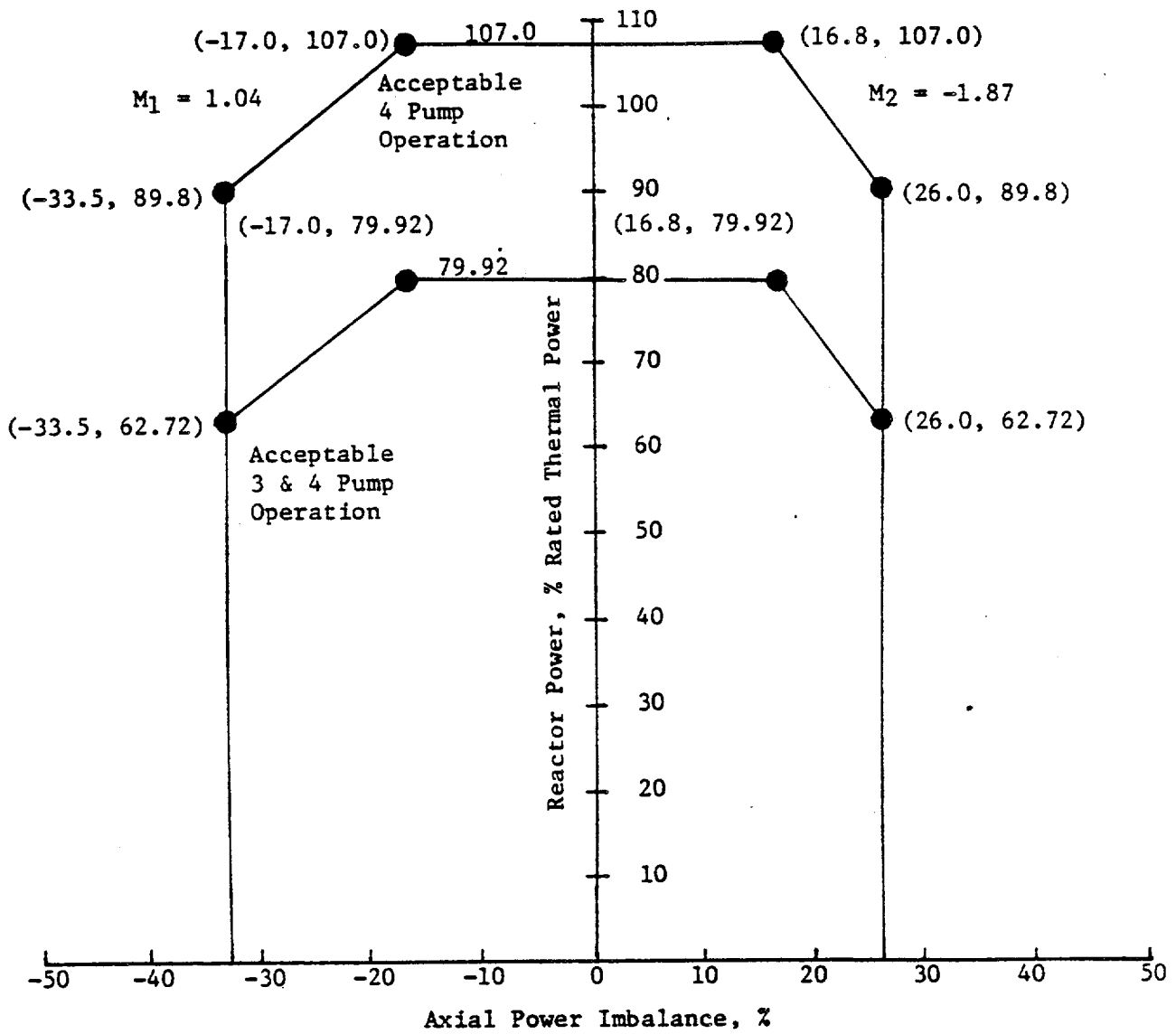
<u>FUNCTION UNIT</u>	<u>TRIP SETPOINT</u>	<u>ALLOWABLE VALUES</u>
8. Pump Status Based on Reactor Coolant Pump Power Monitors (1)	More than one pump drawing ≤ 1152 or $\geq 14,400$ kw	More than one pump drawing ≤ 1152 or $\geq 14,400$ kw
9. Reactor Containment Vessel Pressure High	≤ 4 psig	≤ 4 psig

2-6

- (1) Trip may be manually bypassed when RCS pressure ≤ 1720 psig by actuating Shutdown Bypass provided that:
- The Nuclear Overpower Trip Setpoint is $\leq 5\%$ of RATED THERMAL POWER
 - The Shutdown Bypass RCS Pressure - High Trip Setpoint of ≤ 1720 psig is imposed, and
 - 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



SAFETY LIMITS

BASES

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 22% for that particular reactor coolant pump situation. The 1.30 DNBR curve for three pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the three 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 less than or equal to 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 104.9% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112% which was used in the safety analysis.

LIMITING SAFETY SYSTEM SETTINGS

BASES

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip less than or equal to 618°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 greater than or equal to 107% and reactor flow rate is 100%, or flow rate is less than or equal to 93.45% and power level is 100%.
2. Trip would occur when three reactor coolant pumps are operating if power is greater than or equal to 79.92% and reactor flow rate is 74.7%, or flow rate is less than or equal to 70.09% and power is 75%.

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

LIMITING SAFETY SYSTEM SETTINGS

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 1.07% 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, 2300 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 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, $(11.59 T_{out}^{\circ F} - 5037.8)$ 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 DNS correlation limits, protecting against DNB.

Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of $(11.59 T_{out}^{\circ F} - 5077.8)$ psig.

LIMITING SAFETY SYSTEM SETTINGS

BASES

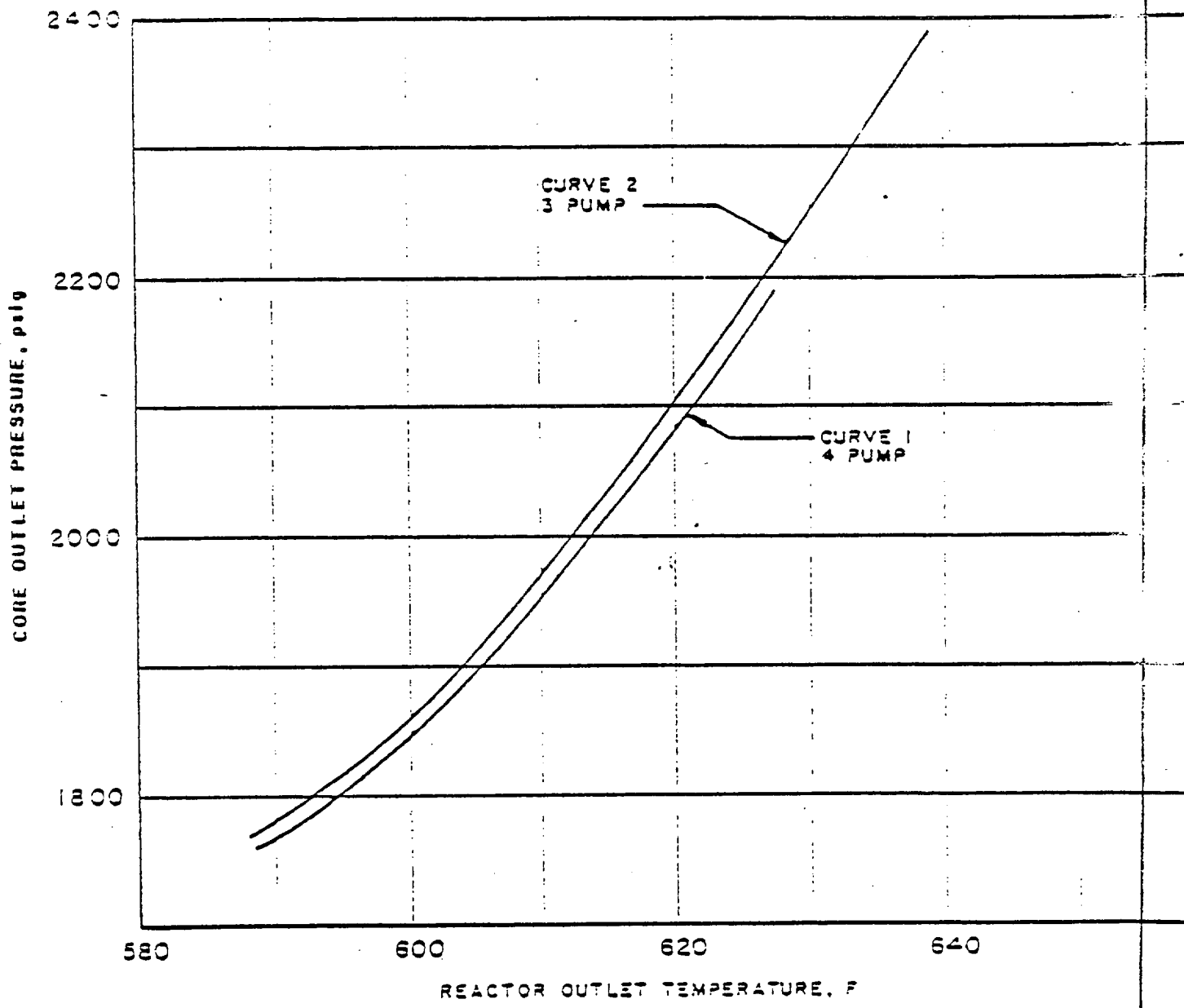
Reactor Containment Vessel Pressure - High

The Reactor Containment Vessel Pressure-High Trip Setpoint ≤ 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 Pump Power Monitors

In conjunction with the power/imbalance/flow trips, the Reactor Coolant Pump Power Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to more than one reactor coolant pump not operating.

A reactor coolant pump is considered to be not operating when the power required by the pump is greater than or equal to 262% (14,400 kw) or is less than or equal to 20.9% (1152 kw) of the operating power (5500 kw). In order to avoid spurious trips during normal operation, the trip setpoints have been selected to maximize the operating band while assuring that a reactor trip will occur upon loss of power to the pump. The 20.9% trip setpoint and response time are based on the maximum time within which an RCPPM-RPS trip must occur to provide DNBR protection for the four pump coastdown. Florida Power has agreed to take credit for the pump overpower trip in order to assure that certain potential faults (such as a seismically induced fault high signal) will not prevent this instrumentation from providing the protective action (i.e. a trip signal). Thus, the maximum setting, approximately 262% (14,400 kw), was selected.



REACTOR COOLANT FLOW			
<u>CURVE</u>	<u>FLOW (% DESIGN)</u>	<u>POWER (RTP)</u>	<u>PUMPS OPERATING (TYPE OF LIMIT)</u>
1	139.7×10^6 (106.5%)	113.05 %	4 PUMPS (DNBR)
2	104.4×10^6 (79.6%)	90.84 %	3 PUMPS (DNBR)

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM ALLOWABLE POWER FOR MINIMUM DNBR
 BASES FIGURE 2.1

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be greater than or equal to 1.0% delta k/k.

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

ACTION:

With the SHUTDOWN MARGIN less than 1.0% delta k/k, immediately initiate and continue boration at greater than or equal to 10 gpm of 11,600 ppm boric acid solution 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 greater than or equal to 1.0% 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} greater than or equal to 1.0.

##With K_{eff} less than 1.0.

*See Special Test Exception 3.10.4.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- e. When in MODE 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.

4.1.1.1.1.2 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.

REACTIVITY CONTROL SYSTEMS

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.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 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 greater than or equal to 105°F.
 - b. At least once per 31 days by verifying that each valve (manual, power operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

REACTIVITY CONTROL SYSTEMS

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4.2 At least one makeup pump shall be OPERABLE.

APPLICABILITY: MODE 4*

ACTION:

With no makeup pump OPERABLE, restore at least one makeup pump to OPERABLE status within one hour or be borated to a SHUTDOWN MARGIN equivalent to 1.0% delta k/k at 200°F and be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

*With RCS pressure greater than or equal to 150 psig.

REACTIVITY CONTROL SYSTEMS

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 with 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.

REACTIVITY CONTROL SYSTEMS

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 6,356 gallons,
 2. Between 11,600 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 2,270 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

REACTIVITY CONTROL SYSTEMS

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 $< 40^{\circ}\text{F}$.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.9 Each 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 6,356 gallons,
 2. Between 11,600 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 415,200 gallons,
 2. Between 2,270 and 2,450 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 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 HOT 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 HOT SHUTDOWN within the following 30 hours.

REACTIVITY CONTROL SYSTEMS

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 outside air temperature is less than 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 $< 1.0\% \Delta k$ at zero power and $< 0.65\% \Delta k$ 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.

REACTIVITY CONTROL SYSTEMS

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, 3.1-1a, 3.1-2, 3.1-2a, 3.1-3, 3.1-3a, 3.1-4 and 3.1-4a 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} greater than or equal to 1.0.

REACTIVITY CONTROL SYSTEMS

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
FOUR PUMP OPERATION FROM 0 EFPD TO 30. (+10/-0) EFPD

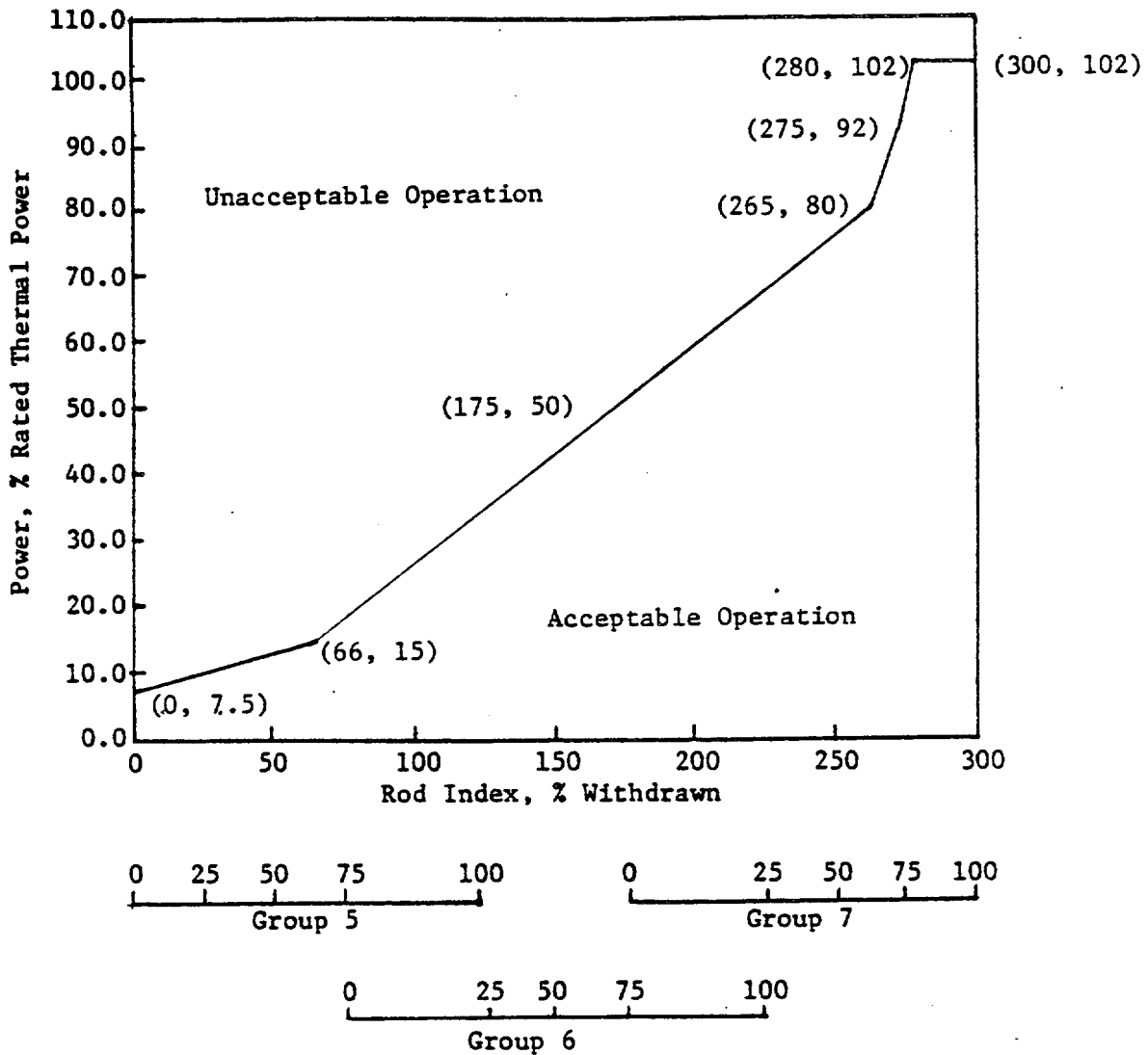


FIGURE 3.1-1a

REGULATING ROD GROUP INSERTION LIMITS FOR
FOUR PUMP OPERATION FROM 30 (+10/-0) TO 250 \pm 10 EFPD

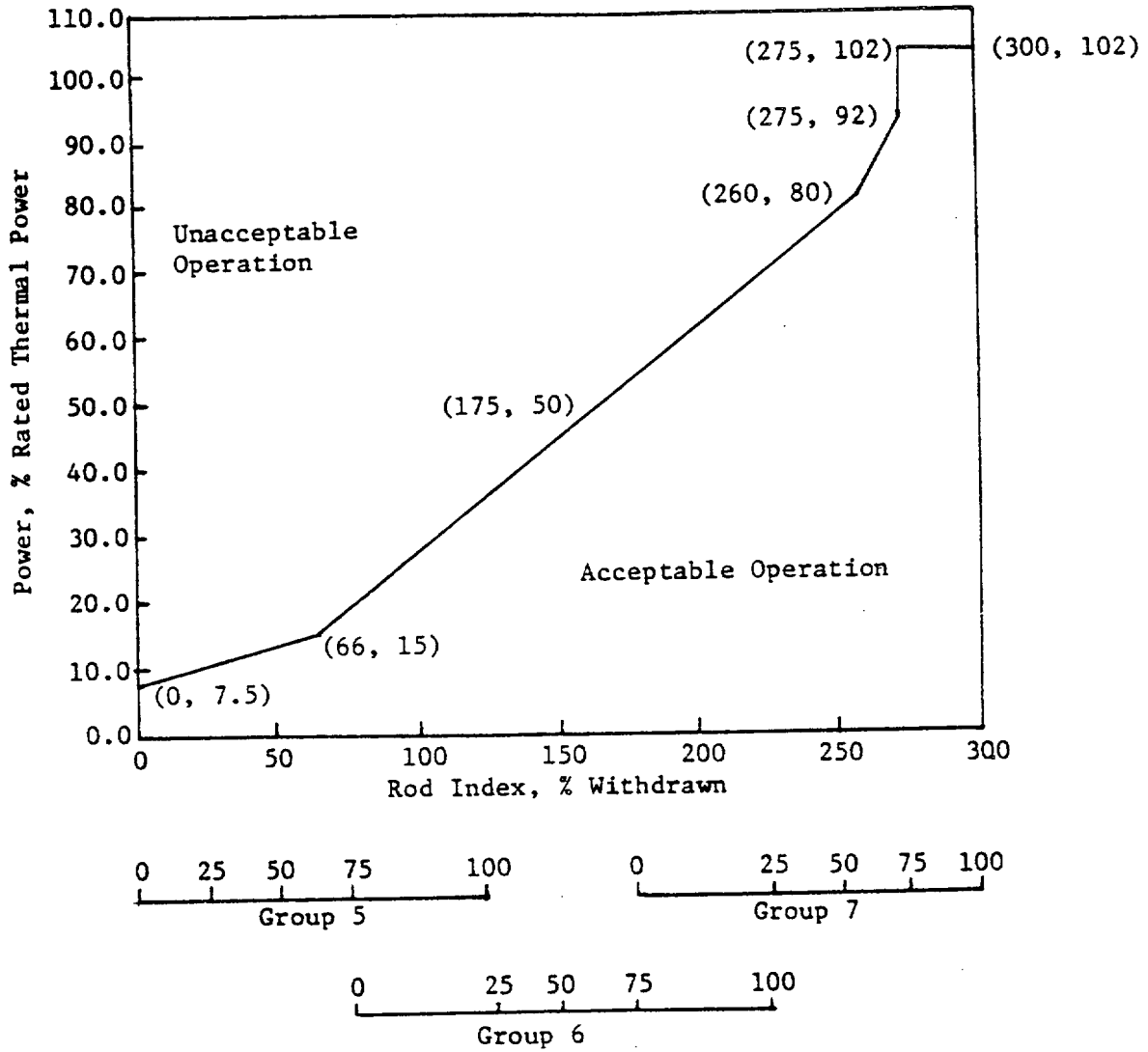


FIGURE 3.1-2

REGULATING ROD GROUP INSERTION LIMITS FOR
FOUR PUMP OPERATION FROM 250 ± 10 TO 399 ± 10 EFPD

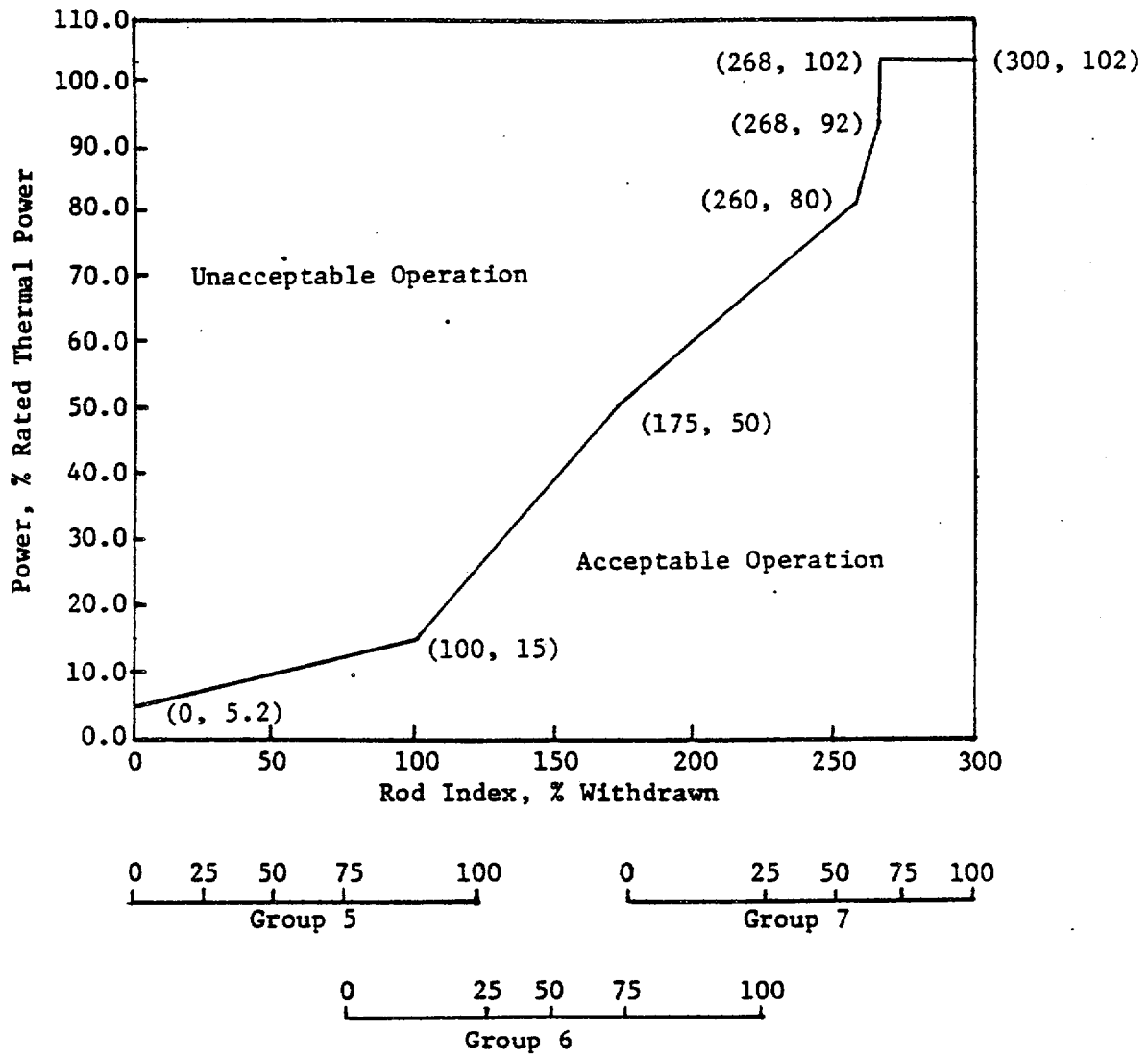


FIGURE 3.1-2a

REGULATING ROD GROUP INSERTION LIMITS FOR
FOUR PUMP OPERATION AFTER 399 ± 10 EFPD

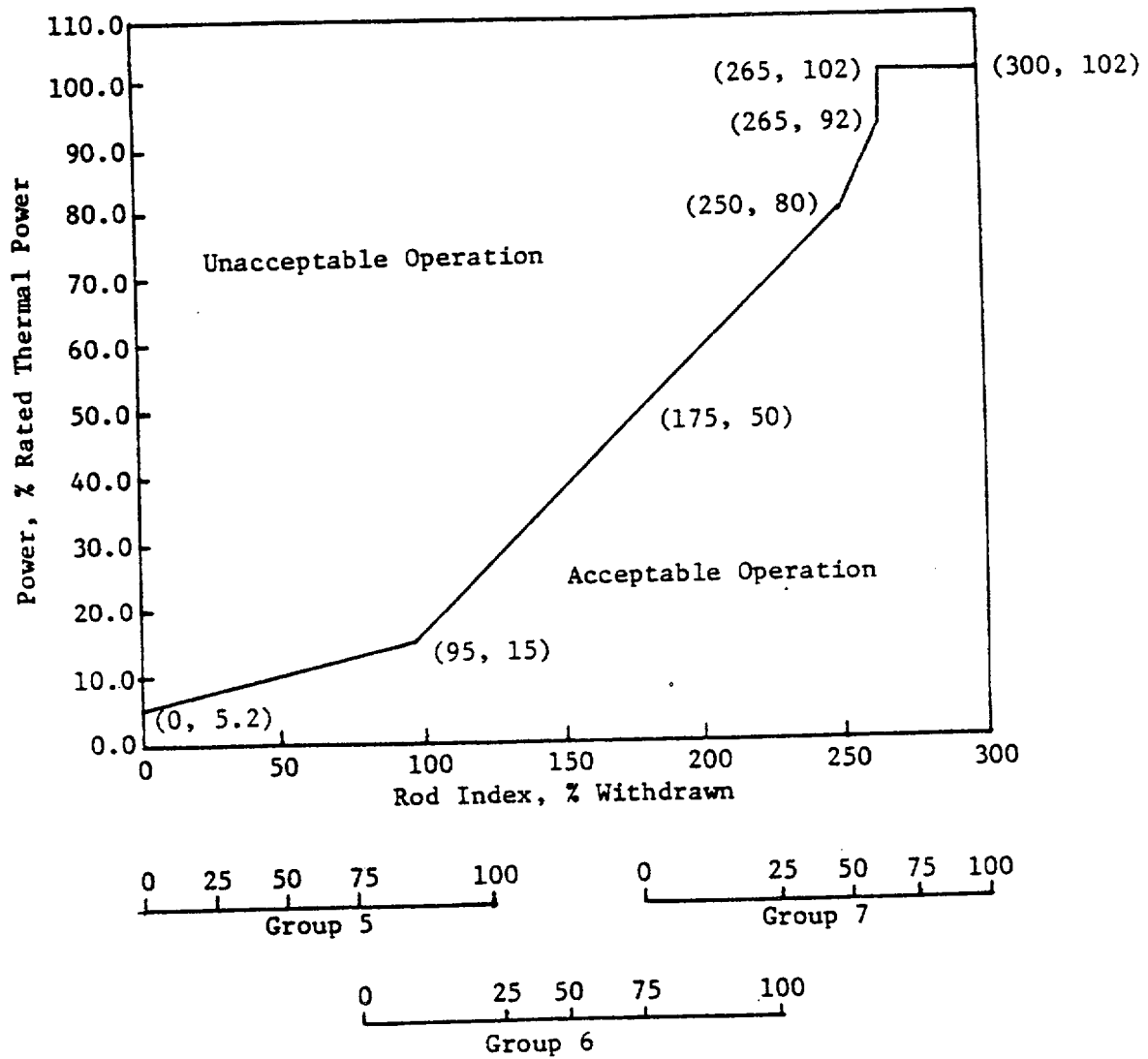


FIGURE 3.1-3

REGULATING ROD GROUP INSERTION LIMITS FOR
THREE PUMP OPERATION FROM 0 TO 30 (+10/-0) EFPD

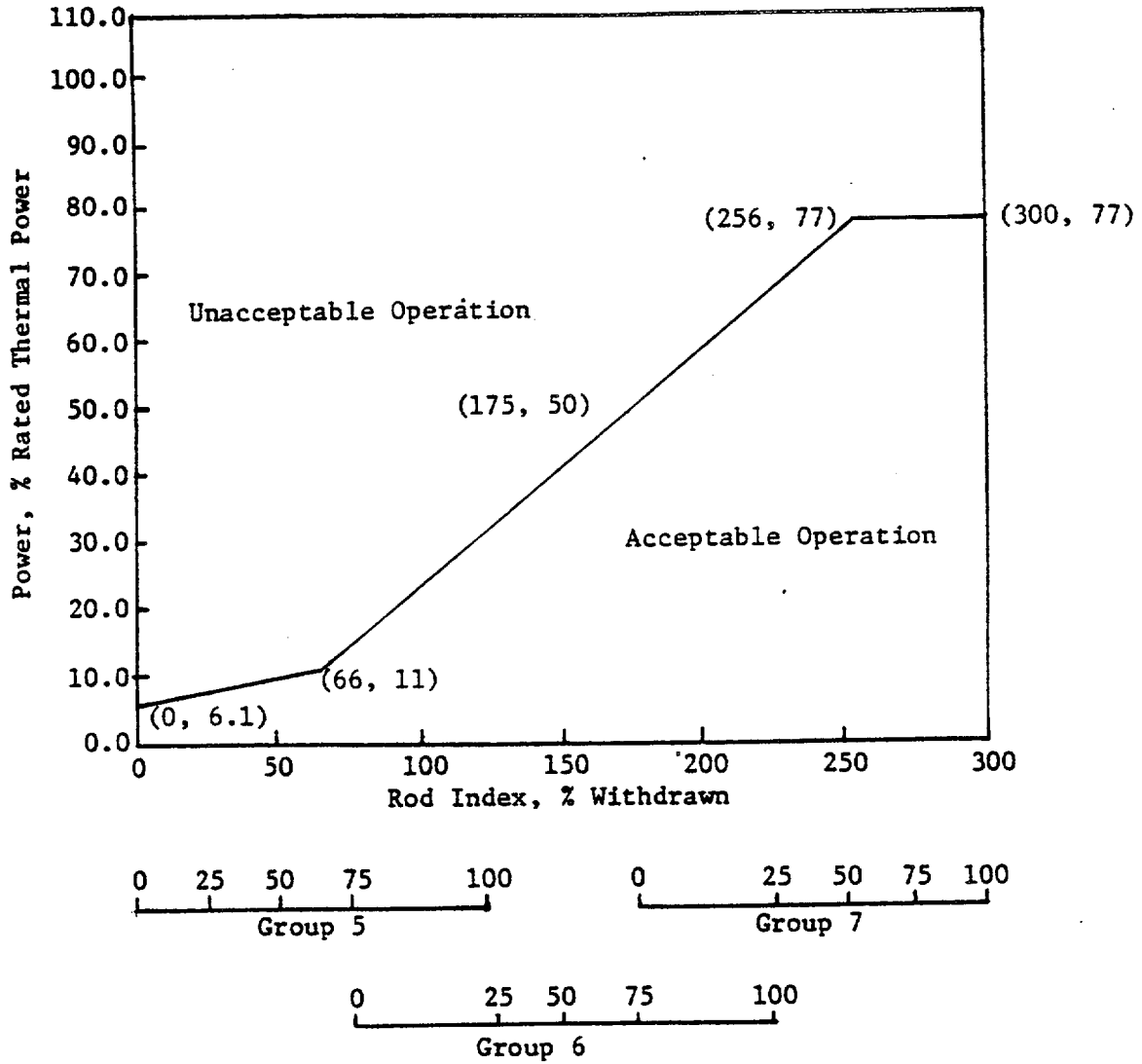


FIGURE 3.1-3a

REGULATING ROD GROUP INSERTION LIMITS FOR
THREE PUMP OPERATION FROM 30 (+10/-0) TO 250 ± 10 EFPD

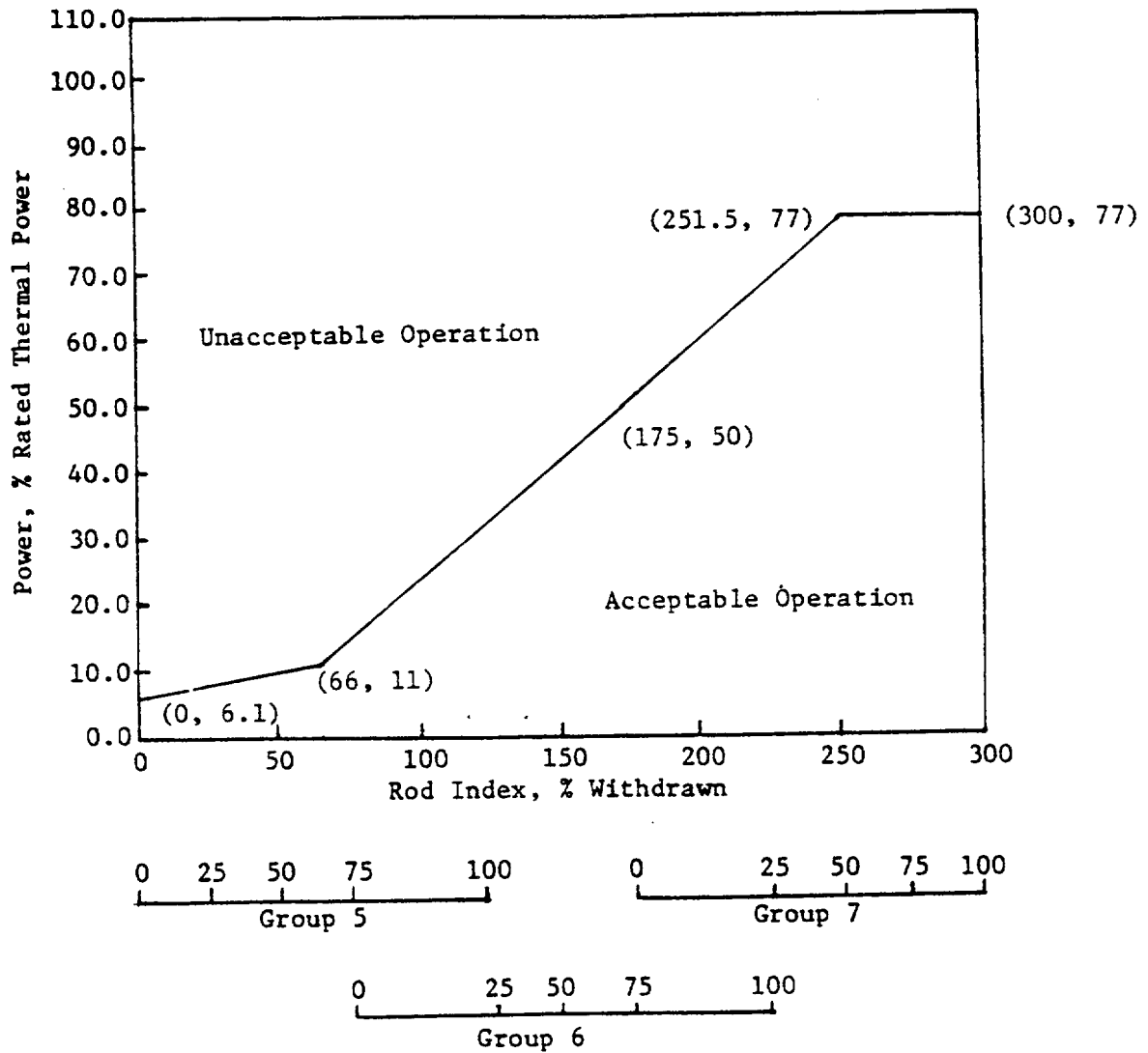


FIGURE 3.1-4

REGULATING ROD GROUP INSERTION LIMITS FOR
THREE PUMP OPERATION FROM 250 ± 10 TO 399 ± 10 EFPD

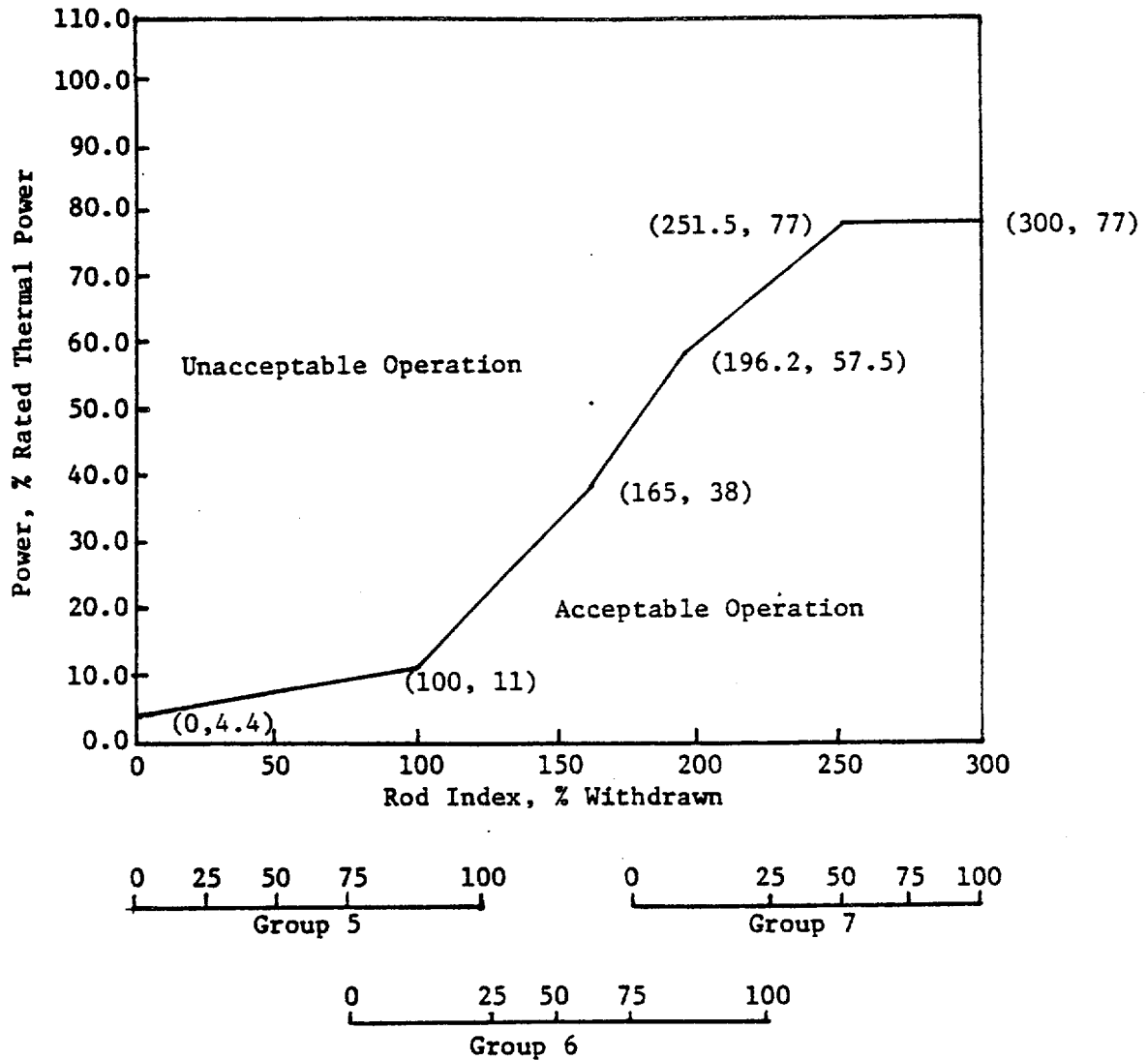
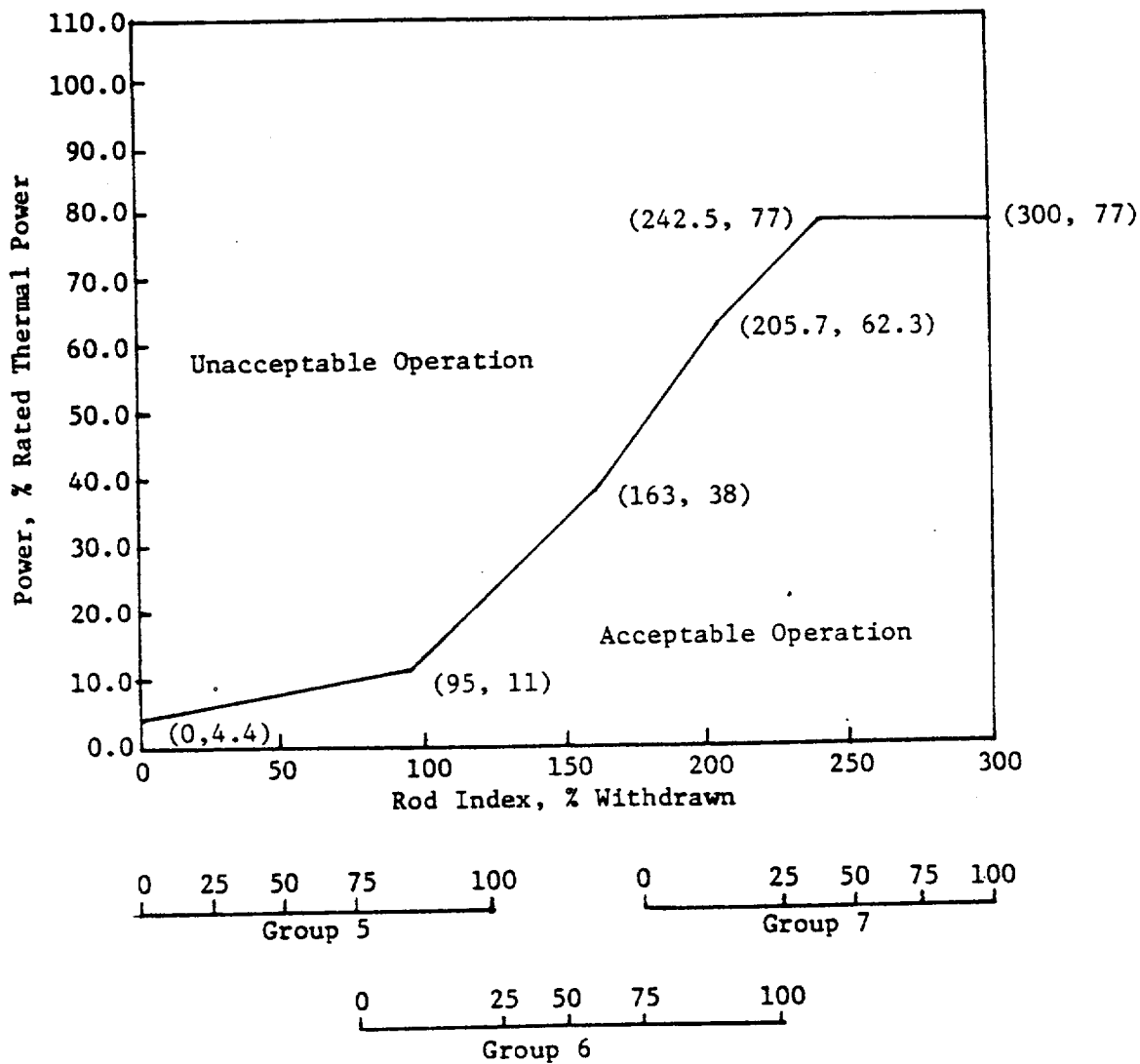


FIGURE 3.1-4a

REGULATING ROD GROUP INSERTION LIMITS FOR
THREE PUMP OPERATION AFTER 399 ± 10 EFPD



DELETED

REACTIVITY CONTROL SYSTEMS

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.

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

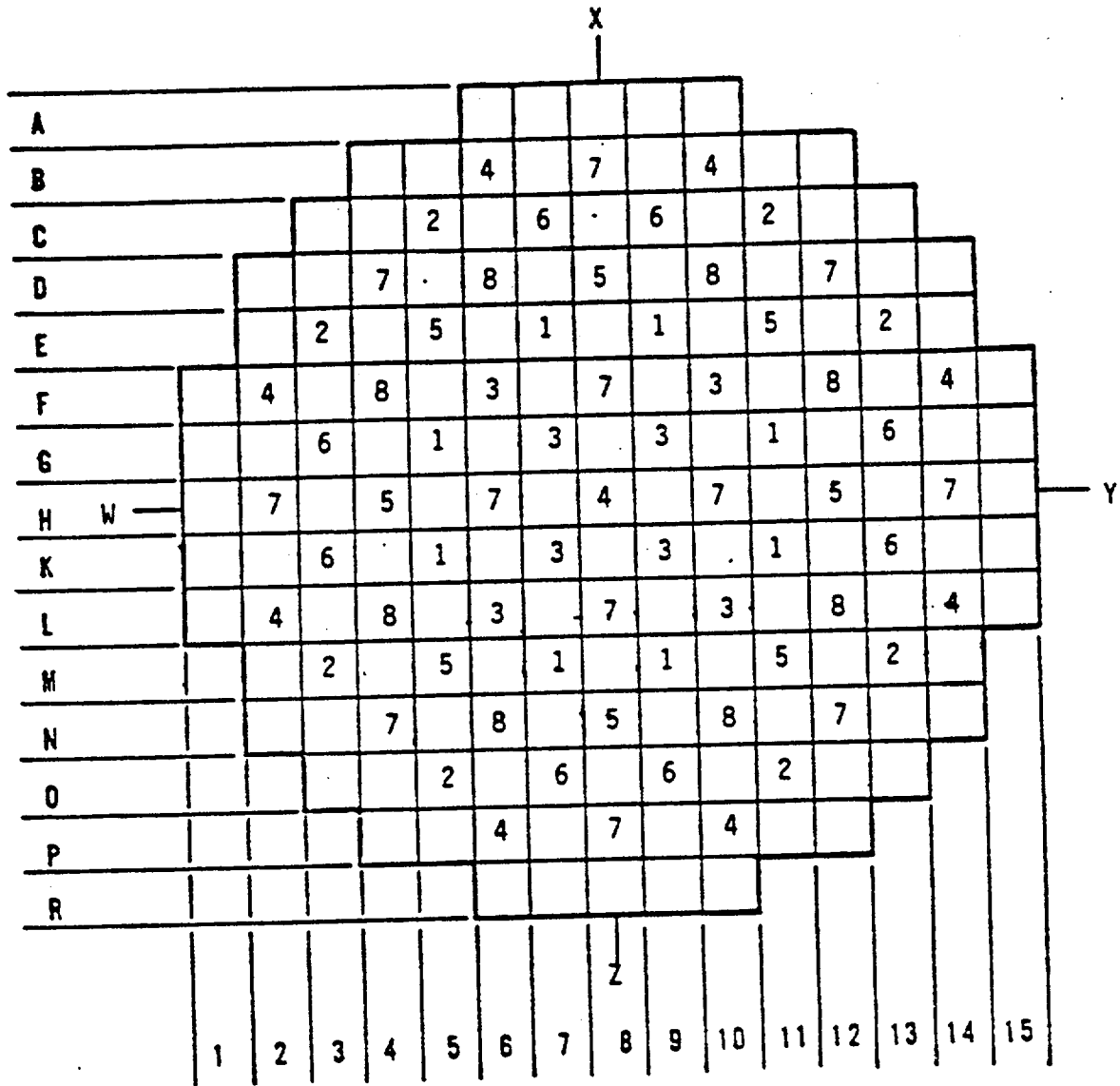
4.1.3.7

- 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 patches 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.

FIGURE 3.1-7

CONTROL ROD LOCATIONS AND GROUP DESIGNATIONS
FOR CRYSTAL RIVER 3, CYCLE 5



x Group Number

Group	No. of Rods	Function
1	8	Safety
2	8	Safety
3	8	Safety
4	9	Safety
5	8	Control
6	8	Control
7	12	Control
8	8	APSRs

Total 69

REACTIVITY CONTROL SYSTEMS

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.9 The axial power shaping rod group shall be limited in physical insertion as shown on Figures 3.1-9, 3.1-9a, and 3.1-10.

APPLICABILITY: MODES 1 and 2*.

ACTION:

With the axial power shaping rod group outside the above insertion limits, either:

- a. Restore the axial power shaping rod group 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 figure within 2 hours, or
- c. Be in at least HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.9 The position of the axial power shaping rod group shall be determined to be within the insertion limits at least once every 12 hours.

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

FIGURE 3.1-9

AXIAL POWER SHAPING ROD GROUP INSERTION
LIMITS FROM 0 TO 30 (+10/-0) EFPD

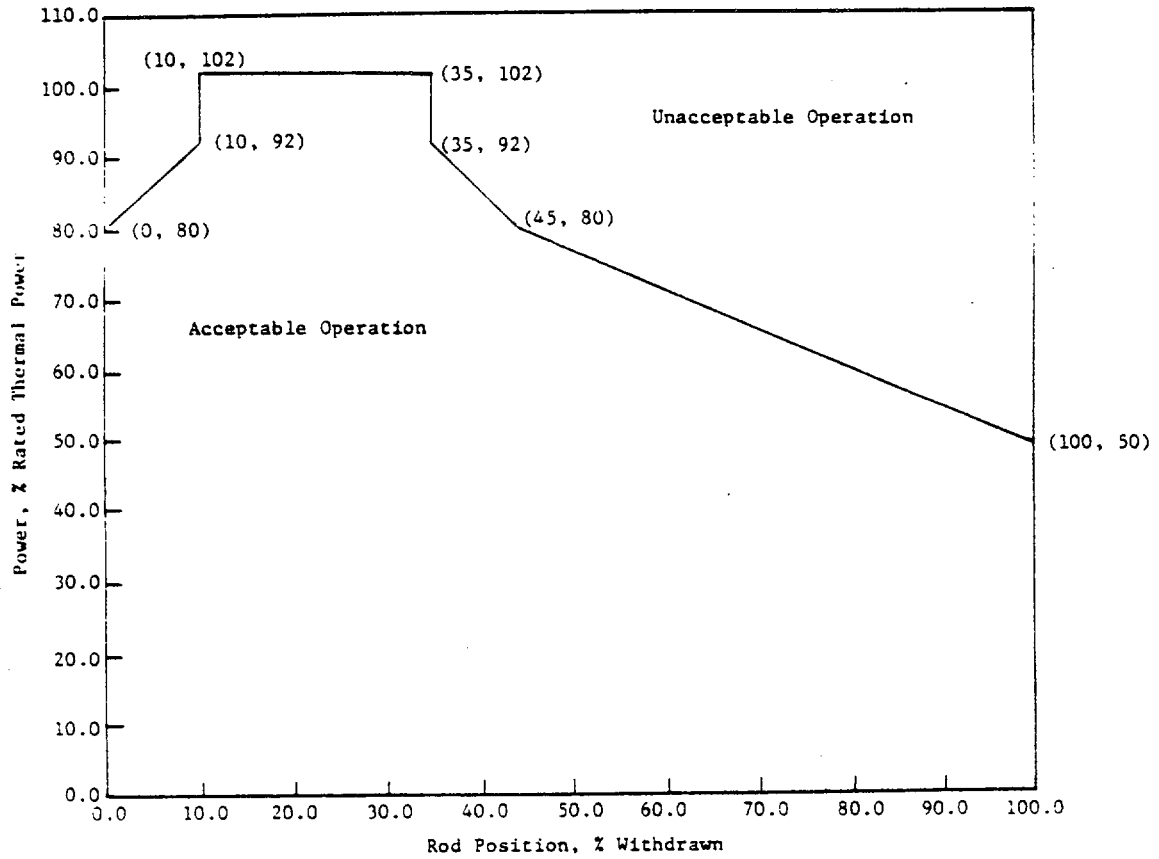


FIGURE 3.1-9a

AXIAL POWER SHAPING ROD GROUP INSERTION
LIMITS FOR 30 (+10/-0) TO 250 ± 10 EFPD

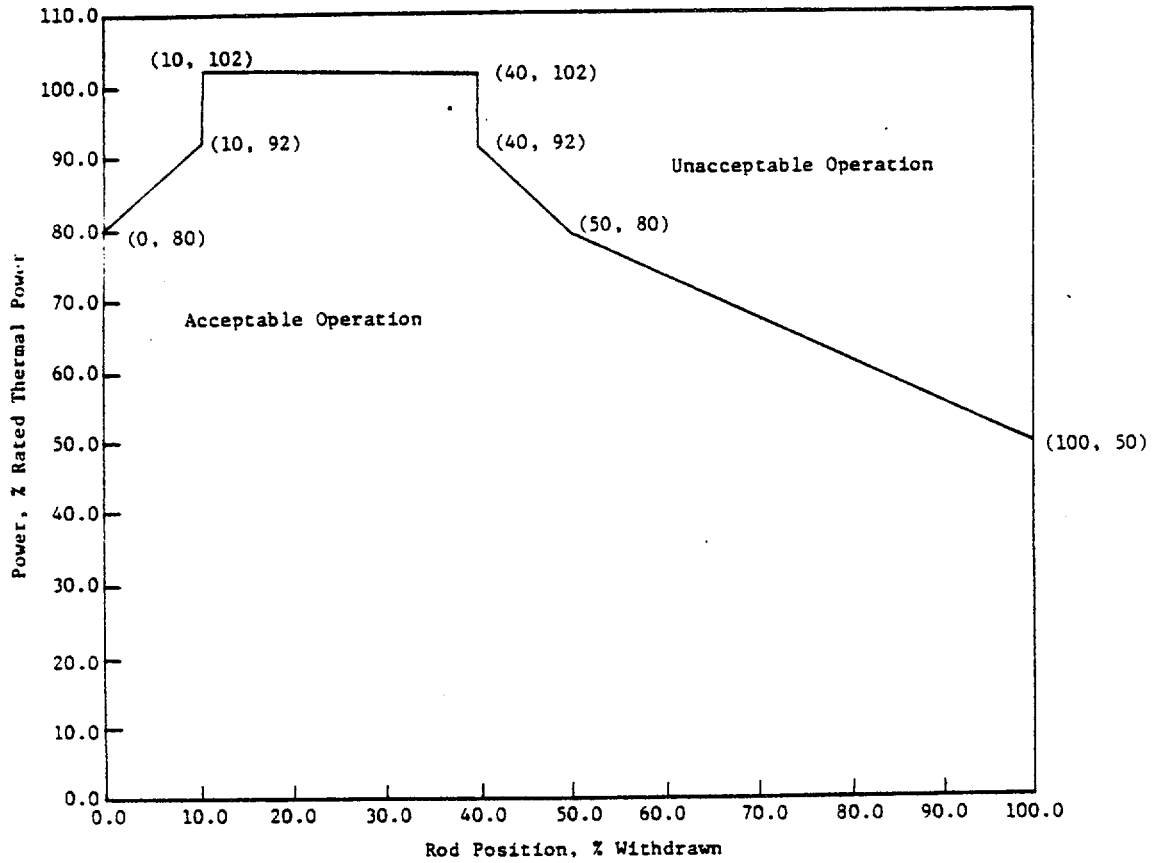
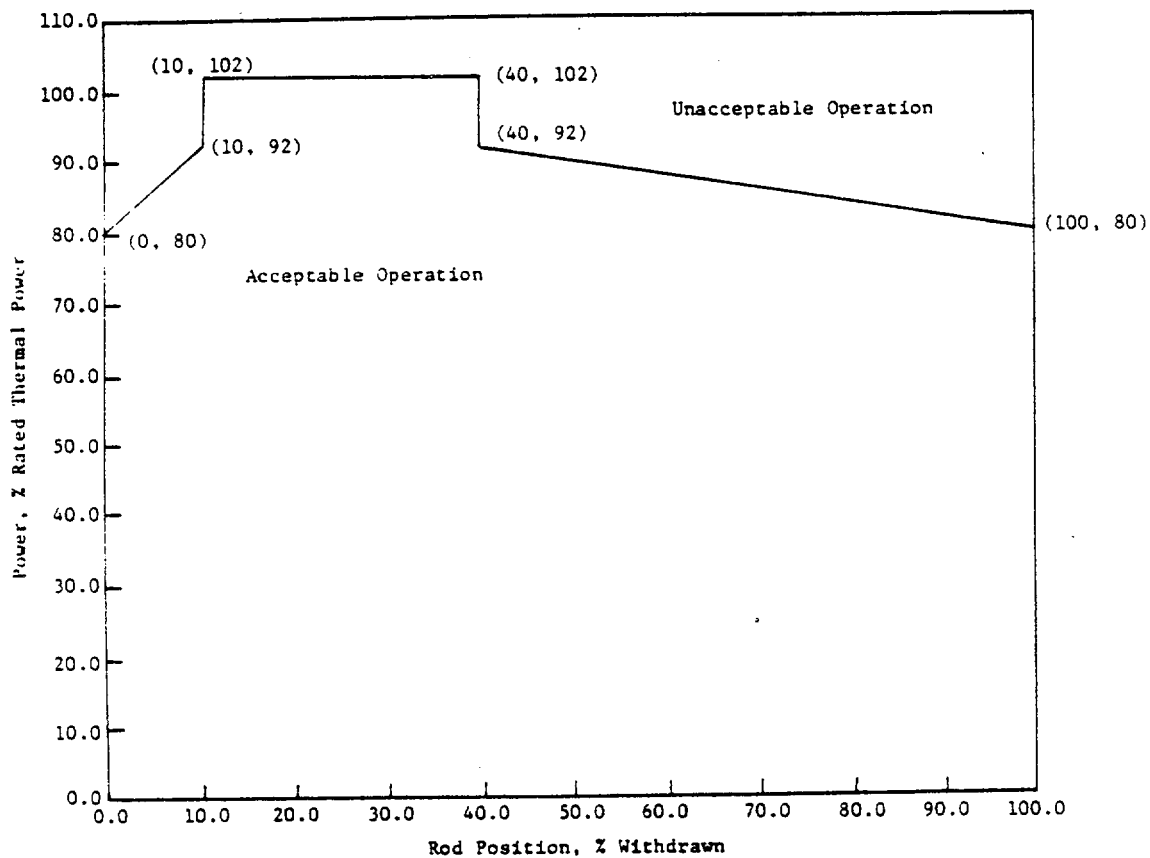


FIGURE 3.1-10

AXIAL POWER SHAPING ROD GROUP INSERTION
LIMITS FOR 250 ± 10 TO 399 ± 10 EFPD



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, 3.2-1a, 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.

*See Special Test Exception 3.10.1.

FIGURE 3.2-1

AXIAL POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 0 TO 30 (+10/-0) EFPD

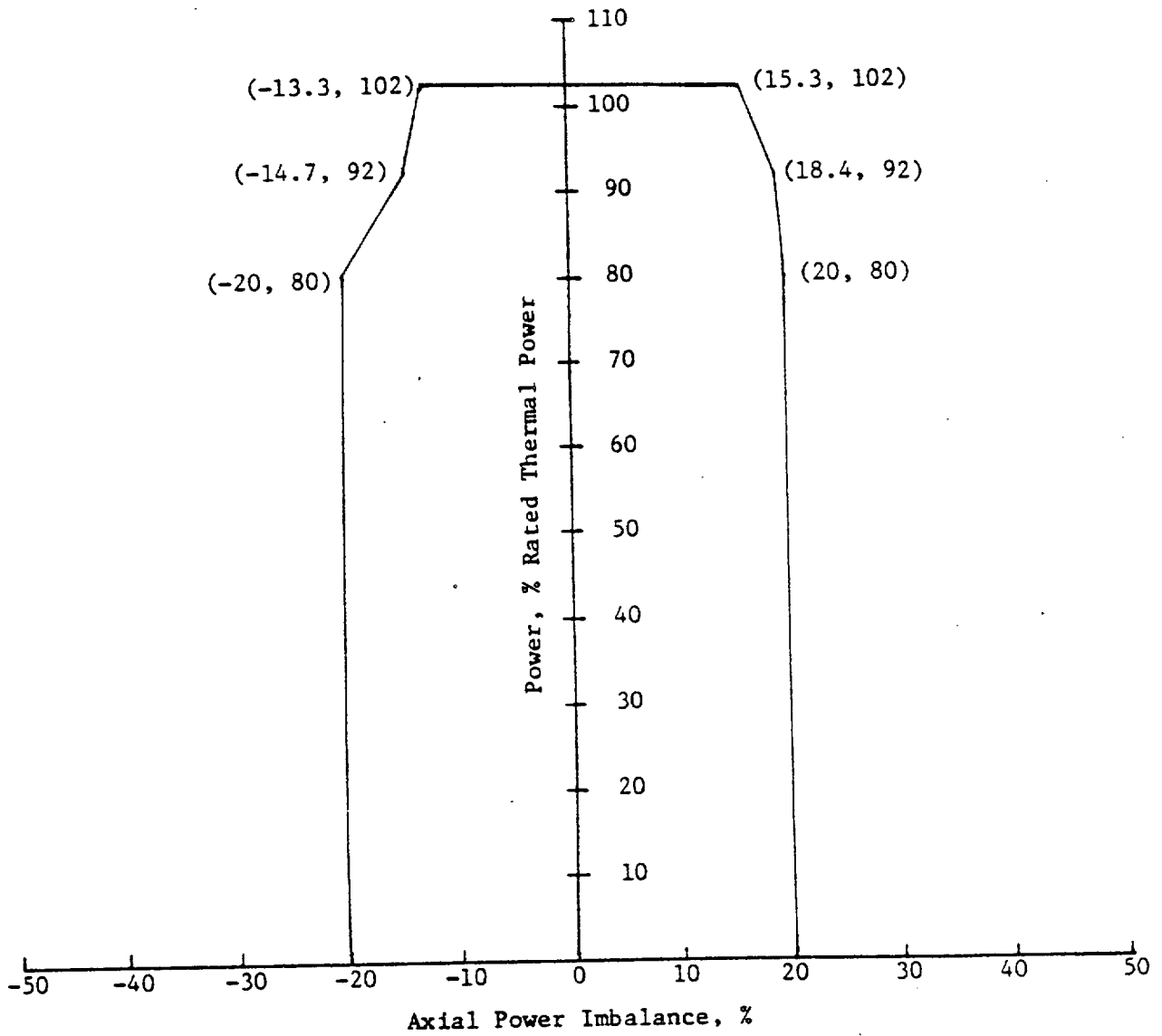


FIGURE 3.2-1a

AXIAL POWER IMBALANCE ENVELOPE FOR
OPERATION FROM 30 (+10/-0) TO 250 \pm 10 EFPD

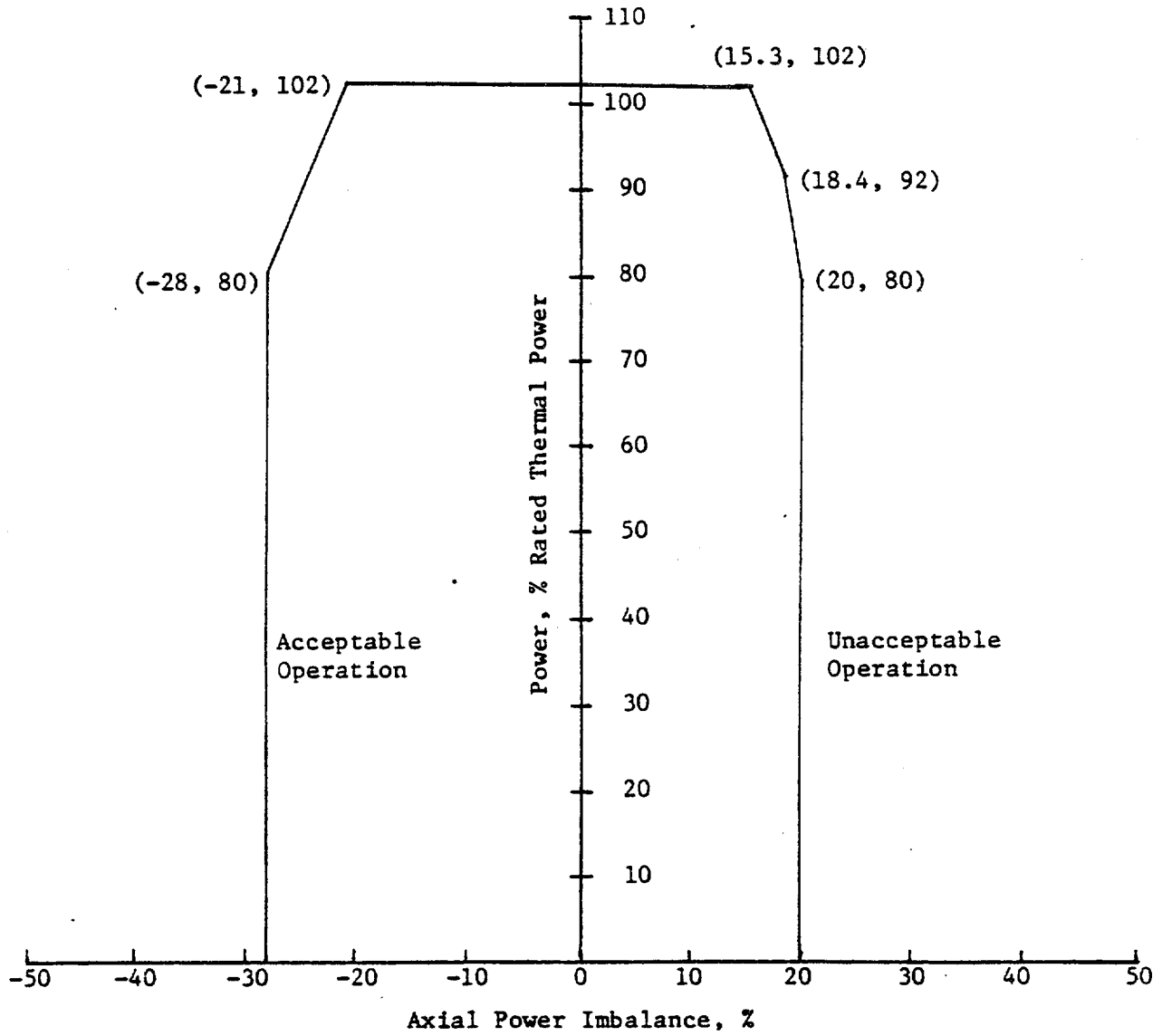
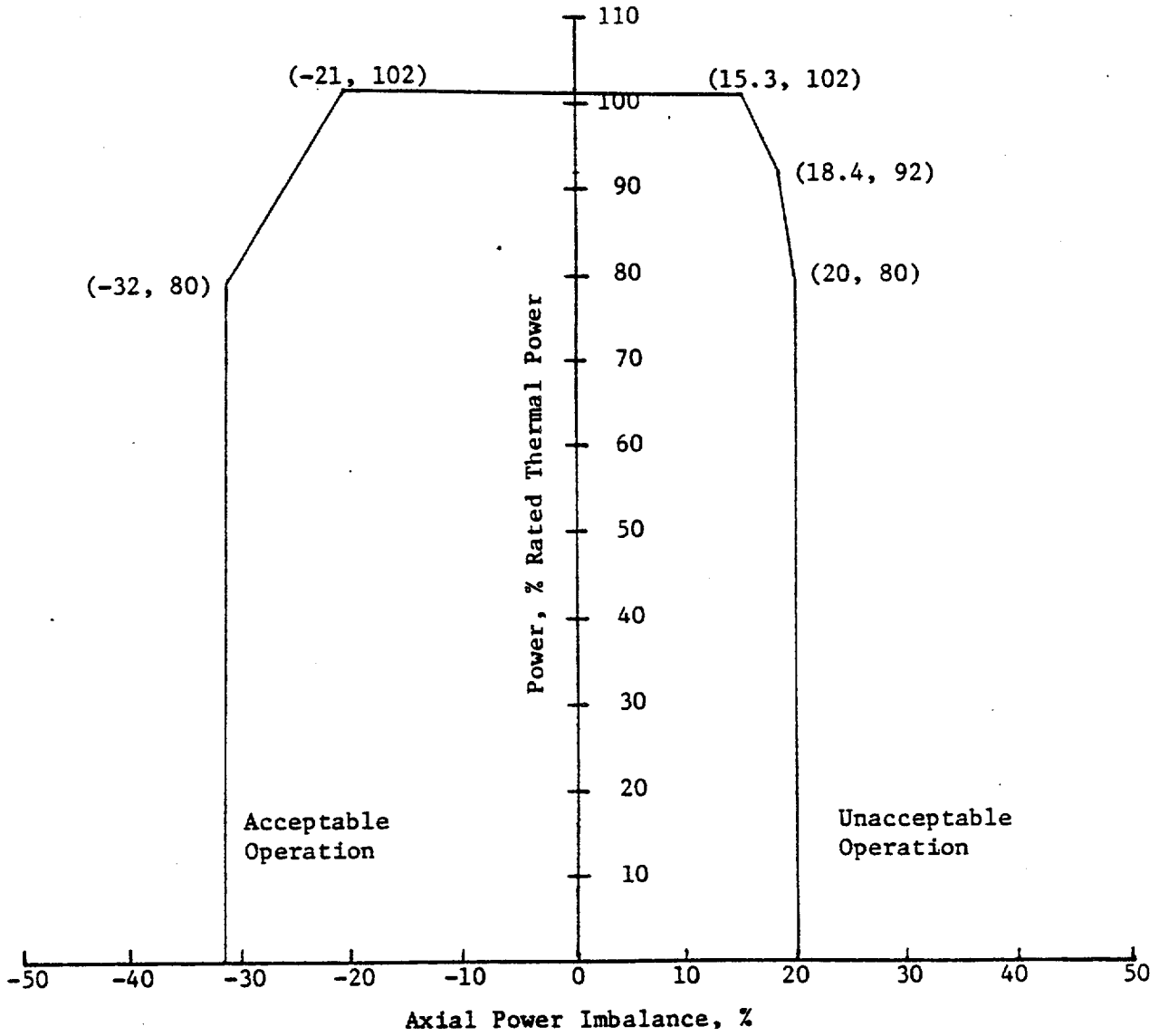


FIGURE 3.2-2

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION AFTER 250 ± 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:

$$F_Q \leq \frac{3.08}{P}$$

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 8% 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 80% 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:

- ACTION 5 - With \leq number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL POWER level:
- $\leq 10^{-10}$ 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.
 - $> 10^{-10}$ 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:
- Within 1 hour:
 - Place the inoperable channel in the tripped condition, or
 - Remove power supplied to the control rod trip device associated with the inoperative channel.
 - 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. 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.
- ACTION 25 - With the number of channels OPERABLE one less than the required Minimum Channels OPERABLE requirement, plant operation may continue until the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 4 hours.

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

Functional Unit	Response Times
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower *	≤ 0.266 seconds
3. RCS Outlet Temperature - High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE *	≤ 1.79 seconds
5. RCS Pressure - Low	≤ 0.44 seconds
6. RCS Pressure - High	≤ 0.44 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Pump Status Based on RCPPIs**	≤ 1.44 seconds
9. 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.

** Time response testing of the RCPPIs may exclude testing of the current and voltage sensors and the watt transducer.

CRYSTAL RIVER UNIT 3

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Amendment No. 47, 48, 55, 64

REACTOR COOLANT SYSTEM

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.2 The power operated relief valve (PORV) and its associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

- a. With the PORV inoperable, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With the block valve inoperable, within 1 hour either restore the block valve to OPERABLE status or close the block valve and remove power from the block valve or close the PORV and remove power from the associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- c. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specifications 4.0.5, the PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2.2 The block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

CONTAINMENT SYSTEMS

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.

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 12,970 and 13,920 gallons of solution containing between 60,000 and 75,000 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

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 the flow rate in the spray additive system.

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.

CRYSTAL RIVER - UNIT 3

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TABLE 4.7-1STEAM LINE SAFETY VALVES

<u>VALVE NUMBER</u>	<u>LIFT SETTING ($\pm 1\%$) (psig)</u>	<u>ORIFICE SIZE (inches)</u>
<u>STEAM GENERATOR 3A</u>		
<u>Main steam line A1</u>		
MSV - 34	1050	4.515
MSV - 38	1070	4.515
MSV - 43	1090	4.515
MSV - 40	1100	3.750
<u>Main steam line A2</u>		
MSV - 33	1050	4.515
MSV - 37	1070	4.515
MSV - 42 ✓	1090	4.515
MSV - 46	1100	4.515
<u>STEAM GENERATOR 3B</u>		
<u>Main steam line B1</u>		
MSV - 35	1050	4.515
MSV - 39 ✓	1070	4.515
MSV - 44 ✓	1090	4.515
MSV - 47	1100	4.515
<u>Main steam line B2</u>		
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 pump and/or associated flow path 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 the steam turbine driven pump develops a discharge pressure greater than or equal to 1100 psig on recirculation flow when the secondary steam supply pressure is greater than 200 psig.*
 2. Verifying that the motor driven pump develops a discharge pressure of greater than or equal to 1100 psig on recirculation flow.

* When not in MODES 1, 2, or 3, surveillance shall be performed within 24 hours after entering MODE 3 and prior to entering MODE 2.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that each valve in the flow path is in its correct position.
 4. Verifying that the emergency feedwater ultrasonic flow rate detector is zero-checked.
- 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 the steam turbine driven pump and the motor driven pump start automatically:
 - a. Upon receipt of an emergency feedwater actuation OTSG A and B level low-low test signal, and
 - b. Upon receipt of an emergency feedwater actuation main feedwater pump turbines A and B control oil low test signal.
 3. Verifying that the operating air accumulators for FWV-39 and FWV-40 maintain greater than or equal to 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.

TABLE 4.7-4

HYDRAULIC SNUBBER INSPECTION SCHEDULE

<u>NUMBER OF SNUBBERS FOUND INOPERABLE DURING INSPECTION OR DURING INSPECTION INTERVAL*</u>	<u>NEXT REQUIRED INSPECTION INTERVAL**</u>
0	18 months \pm 25%
1	12 months \pm 25%
2	6 months \pm 25%
3 or 4	124 days \pm 25%
5, 6, or 7	62 days \pm 25%
Greater than or equal to 8	31 days \pm 25%

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

** The required inspection interval shall not be lengthened more than one step at a time. Following the 1983 refueling outage, the first inservice visual inspection of snubbers shall be performed after 4 months but within 10 months of commencing POWER OPERATION. Subsequent intervals shall be determined by the above Table.

PLANT SYSTEMS

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 ≥ 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 decortaminated 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:

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 for Modes 1, 2, and 3 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 requirements 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.

REACTIVITY CONTROL SYSTEMS

BASES

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 that (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 RTNDT 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% Δ k/k after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs from full power equilibrium xenon conditions and requires either 6356 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 43,478 gallons of 2,270 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 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 1.0% Δ k/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 300 gallons of 11,600 ppm boron from the boric acid storage system or 1,608 gallons of 2,270 ppm boron from the borated water storage tank. To envelop future cycle BWST contained borated water volume requirements, a minimum volume of 13,500 gallons is specified.

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 RELIEF VALVES - SHUTDOWN

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.

3/4.4.3 RELIEF VALVES - OPERATING

The power operated relief valve (PORV) operates to relieve RCS pressure below the setting of the pressurizer code safety valves. This relief valve has a remotely operated block valve to provide a positive shutoff

REACTOR COOLANT SYSTEM

BASES

capability should the PORV become inoperable.

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 trip. The high level limit is based on maximum reactor coolant inventory assumed in the safety analysis.

The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve 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 ensures 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

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 118.3 percent of the total secondary steam flow of 11.0×10^6 lbs/hr at 100% RATED THERMAL POWER.

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:

$$SP = \left[\frac{X - AY}{X^1} \right] \times 105.5$$

where: SP = reduced Nuclear Overpower Trip Setpoint in percent of Rated Thermal Power

X = total actual relieving capacity of each steam generator in lbs/hour (6,503,887 lbs/hour)

A = maximum number of inoperable safety valves per steam generator

Y = maximum relieving capacity of each of the larger capacity safety valves in lbs/hour (845,759 lbs/hour)

X¹ = total required relieving capacity of each steam generator for 112% Rated Thermal Power in lbs/hour (6,160,000 lbs/hour)

105.5 = Nuclear Overpower Trip Setpoint specified in Table 2.2.1

PLANT SYSTEMS

BASES

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.

Each 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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 64 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

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RELOAD SAFETY EVALUATION
CRYSTAL RIVER UNIT 3
FUEL CYCLE 5

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1.0 Introduction

By letter dated March 31, 1983 (Ref. 1), Florida Power Corporation (FPC or the licensee) requested amendment of the Technical Specifications (TSs) of Facility Operating License No. DPR-72 for Crystal River Unit 3 Nuclear Generating Plant to permit operation for a fifth cycle. The safety analyses performed and the resulting modifications to the plant TSs are described in the Cycle 5 reload report (Ref. 2).

The safety analysis for the previous fourth cycle of operation at Crystal River Unit 3 is being used by the licensee as a reference for the proposed fifth cycle of operation. Where conditions are identified as limiting in the fourth cycle analysis, our previous evaluation (Ref. 3) of that cycle continues to apply.

1.1 Description of the Cycle 5 Core

The Crystal River Unit 3 core consists of 177 fuel assemblies, each of which is a 15X15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 5 will operate in a feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length control rod assemblies (CRAs) and 56 burnable poison rod assemblies (BPRAs). In addition, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The licensed core full power level is 2544 Mwt.

2.0 Evaluation of the Fuel System Design

2.1 Fuel Assembly Mechanical Design

The 76 Babcock and Wilcox (B&W) Mark-B4 15X15 fuel assemblies loaded as Batch 7 at the end of Cycle 4 (EOC 4) are mechanically interchangeable with Batches 4D, 5B, 6A, and 6B fuel assemblies loaded previously at Crystal River Unit 3. The Mark-B4 fuel assembly has been previously approved (Ref. 3) by the NRC staff and is utilized in other B&W nuclear steam supply systems. Two assemblies will contain regenerative neutron sources, and retainers (Refs. 4 and 5) will be used to contain these sources as well as the BPRAs. Reinsertion of the BPRAs was approved for the previous cycle of operation, which increased the cycle lifetime to 350 effective full power days (EFPD). The design cycle lifetime of Cycle 5 is 460 EFPD.

A number of recent changes to the B&W 15X15 fuel assembly design (e.g., a larger fuel assembly holddown spring, fuel pellets manufactured by an alternate supplier, combined fixed control component spider and retainer to replace the retainers described above) have been made in other operating B&W 177-fuel-assembly plants and so far these design changes have been found acceptable. If such changes are incorporated into future cycles of operation at Crystal River Unit 3, they should be noted in the appropriate reload safety analysis report. For the current cycle (Cycle 5), however, the licensee has stated that the design of the fresh fuel is identical to that previously approved and irradiated at the plant. The fuel mechanical design is, therefore, acceptable.

2.2 Fuel Rod Design

Although all batches in the Crystal River Unit 3 Cycle 5 core will utilize the same Mark-B4 fuel design and are mechanically interchangeable, the Batch 7 assemblies will incorporate a slightly higher average enrichment. Sixty-eight assemblies will contain 3.29 w/o U-235 and are designated Batch 7A. The eight remaining Batch 7 assemblies will contain 2.95 w/o U-235 and are designated Batch 7B. The 2.95 w/o U-235 enrichment is identical to that in the previously loaded Batch 6B fuel. The fuel rod design parameters will be identical for all assemblies in the Cycle 5 core with the exception of Batch 4D, a single assembly with minor internal differences from other assemblies in the core. The cladding stress, strain and collapse analyses for the Cycle 5 fuel rod design are bounded by conditions previously analyzed for Crystal River Unit 3 or were analyzed specifically for Cycle 5 using methods and limits previously reviewed and approved by the NRC. We find that no further review of these areas is necessary.

2.2.1 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (SRP) (Ref. 6) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those which may affect the operation of the fuel rod is the internal pressure limit. The acceptance criterion (SRP 4.2, Section II.A.1(f)) is that the fuel rod internal gas pressure should remain below normal system pressure unless otherwise justified.

The licensee has stated that the fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 5. This analysis is based on the use of the B&W TAFY-3 code (Ref. 7) rather than one of the newer B&W codes, TACO-1 (Ref. 8) and TACO-2 (Ref. 9). Although all of these codes have been approved for use in safety analysis, we believe (Ref. 10) that only the newer TACO series of codes are capable of correctly calculating fission gas release (and, therefore, rod pressure) at very high burnups. B&W has responded (Ref. 11) to this concern with an analytical comparison between the TAFY-3 and TACO-1 codes. In this response, they have stated that the fuel rod internal pressure predicted by TACO-1 is lower than that predicted by TAFY-3 for fuel rod exposures of up to 42,000 MWd/MtU. Although we have not examined this comparison, we note that the analyses exceed the maximum expected exposure (33,546 MWd/MtU) for all fuel rods in the Crystal River Unit 3 core at the end of Cycle 5. We conclude that the rod internal pressure limits have been adequately considered for Cycle 5 operation.

2.3 Fuel Thermal Design

The thermal behavior of the Cycle 5 core is virtually identical for all fuel assemblies. The licensee has elected to use a combination of the TAFY-3 and TACO-2 codes to analyze the thermal behavior of the fuel, and the Cycle 5 reload is the first time the more recent TACO-2 code has been used in plant safety analysis since our approval of this code was issued (Ref. 12). In general, the thermal analysis for each assembly has been performed with TACO-2 with the exception of Batch 4D (a single fuel assembly), where the previous TAFY-3 analysis continues to apply. For non-loss-of-coolant accident (LOCA) analysis, only minor differences (in the linear heat rate to centerline melt) from the previous reload safety analysis have resulted. We continue to find these results acceptable.

For the LOCA analysis (Section 7.2 of the reload report), the average fuel temperature as a function of linear heat rate and the lifetime pin pressure data were calculated and the licensee has stated that the conditions used in the generic LOCA analysis are conservative compared with those calculated for Cycle 5 at Crystal River Unit 3.

Although B&W currently has several approved fuel performance codes which could be used to calculate LOCA initial conditions, the older TAFY-3 code was used for the generic LOCA analysis cited in the Crystal River Unit 3 Cycle 5 reload report. Information obtained by the NRC staff (Ref. 13) indicates that the TAFY-3 code predictions do not produce higher calculated peak cladding temperatures in the generic LOCA analysis than the newer TACO-1 or TACO-2 codes as suggested by the licensee. The issue involves excessive fuel densification and lowered fuel rod internal gas pressures at beginning of life. B&W has proposed a method of resolving this issue which has been adopted by the licensee (Ref. 14). The method relies on reduced peak linear heat rate (PLHR) limits at low core elevations for the first 30 EFPD of operation based on comparison of TAFY-3 and TACO-2 calculated LOCA initial conditions. The method is similar to an older TAFY-3/TACO-1 comparison (Ref. 15) used in the Crystal River Unit 3 Cycle 4 safety analysis. However, the resulting PLHR reduction is different for each code.

In addition to the issue of initial fuel temperatures and rod internal pressures used in the LOCA analysis, a second issue involving cladding swelling and rupture models has affected the proposed Cycle 5 operating limit for Crystal River Unit 3. On November 1, 1979, the NRC staff met with fuel vendors and industry representatives to discuss these models. The staff presented new models (Ref. 16) that we believed met the requirements of 10 CFR 50 Appendix K. Each fuel vendor was then asked to show how, in light of the new models, the plants analyzed with their analytical methods continued to meet the applicable LOCA limits. The B&W response (Ref. 17) concluded that the impact of the NRC models was small and did not result in analytical results in excess of the LOCA limits.

A more recent B&W calculation (Ref. 18), however, found that the cladding swelling and rupture models presented by the staff had a significant effect on LOCA peak cladding temperatures in B&W 177 fuel assembly plants. Because this calculation was applicable to all B&W plants, the licensee was requested (Ref. 19) to provide supplemental calculations for Crystal River Unit 3 similar

to those provided in Reference 18. The licensee's responses (Refs. 20, 21, 22, and 23) culminated in the supplemental calculation (Ref. 14) previously cited as the source of the TAFY-3/TACO-2 PLHR penalty.

The supplemental calculation was performed for the postulated LOCA in a B&W 177 fuel assembly lowered-loop plant such as Crystal River Unit 3. The most limiting conditions for that event were identified as resulting from a double-ended break of the reactor coolant pump discharge pipe (8.55 square foot break) occurring near beginning of cycle. Limiting cladding temperatures were calculated to occur at the 2-foot core elevation. No other elevations were examined because previous B&W analyses have shown that the LOCA limits at the lower core elevations are limited by the time of rupture and the rupture node temperature. Since the NRC cladding models impact mainly the rupture node cladding temperature, the LOCA limits at the higher core elevations were not expected to be affected more than the LOCA limit at the 2-foot elevation. Above the core midplane (i.e., 8- and 10-foot elevations), the analysis is limited by the unruptured node temperature and not greatly affected by the staff cladding models. As a result, the supplemental calculation assigns the 0.5 kW/ft reduction in PLHR determined at the 2-foot elevation to the 4 and 6-foot elevations as well. This reduction is incurred in addition to the TAFY-3/TACO-2 penalty discussed previously.

In general, the supplemental calculation utilizes previously approved methods except for the substitution of the NRC cladding models. However, there are segments of the analysis (e.g., THETA1-B - Ref. 24) that are currently undergoing NRC review. In addition, the calculations were not performed in an integral manner because of code incoherencies between the unreviewed and previously approved models. We recognize, however, that the calculation results in plant operating conditions which are more restrictive than those previously used at Crystal River Unit 3. The licensee has incorporated these results into the Crystal River Unit 3 TSs to support operation during Cycle 5. These changes have been reviewed by the NRC staff and have been found acceptable. We, therefore, conclude, on an interim basis, that the supplemental calculations provide an acceptable basis for continued operation at Crystal River Unit 3 and that the postulated LOCA has been appropriately considered for Cycle 5 operation.

2.4 Operating Experience

B&W has accumulated operating experience with the Mark B 15X15 fuel assembly at all of the eight operating B&W 177-fuel assembly plants. A summary of this operating experience as of May 31, 1982 is given on page 4-3 of Reference 2.

2.4.1 Iodine Spiking

Our review of the Cycle 2 and Cycle 3 operation at Crystal River Unit 3 made us aware of a substantial number of licensee event reports issued as a result of iodine spiking. Each event has been associated with "known leaking fuel pins." In view of the possible generic problems (in either fuel design or operating limits) associated with such fuel failures, we requested (Ref. 25) additional information regarding these iodine spiking events. The licensee responded (Ref. 26) to our query with a report on iodine spiking at Crystal River Unit 3 covering Cycle 1 through Cycle 4. As stated in the transmittal letter for the

licensee's report, it is the licensee's conclusion that "the level of [fuel] defects is well within the range of that associated with good fuel performance as compared with published industry data. The substantial number of licensee event reports resulted from a combination of iodine spiking, and an overly stringent TS requirement for reporting."

We have not yet completed our review of the licensee's report. However, based on our examination to date, a number of comments can be made. First, the apparent confusion between Dose Equivalent (DE) I-131 in the plant TSs and the isotopic I-131 concentration in the licensee's model casts some doubts on the results presented. Second, the use of a single escape rate coefficient, because of non-linearity effects, is a poor approximation. Third, reference to the licensee's TS primary coolant activity limit of 1.0 $\mu\text{Ci/g DE}$ as "substantially lower than the level of 3.5 $\mu\text{Ci/g [DE]}$ required at plants similar to Crystal River Unit 3" is somewhat misleading. It is true that one B&W 177-FA plant (ANO-1) has a TS limit of 3.5 $\mu\text{Ci/g DE I-131}$. The range of TS limits on primary coolant activity varies substantially (see Table 14 of Reference 27), and some operating reactors have no limits at all. However, for plants similar to Crystal River Unit 3 (recent PWRs), the majority show closer agreement to the Standard Technical Specification (STS) limit of 1.0 $\mu\text{Ci/g DE}$ (like Crystal River 3) than to the 3.5 $\mu\text{Ci/g DE}$ limit at ANO-1. Rather than permit increases in these limits, there is a generic recommendation (Ref. 28) to impose the STS limit on all plants. This effort has taken a new impetus since the occurrence of a steam generator tube rupture event at the R. E. Ginna facility, as a result of which the NRC staff reduced that plant's primary coolant activity limit from 3.0 $\mu\text{Ci/g DE}$ to 0.2 $\mu\text{Ci/g DE I-131}$.

We note that the licensee has not requested an increase in the coolant activity TS limit at this time. We further note that there have been no Licensee Event Reports issued on this subject since October 1981 and the recent Cycle 4 average coolant activity levels have been significantly lower than those reported for previous cycles of operation at Crystal River Unit 3. Since the licensee continues to use the PWR STS limits and surveillance requirements on the primary coolant activity level and since coolant activity levels have been reduced significantly during the previous cycle of operation, we conclude that the issue of iodine spiking has been adequately addressed for Cycle 5 operation.

2.5 Conclusions

We have reviewed those sections of the reload report for Crystal River Unit 3 Cycle 5 dealing with the fuel system design. We find those portions of the application acceptable.

3.0 Evaluation of the Nuclear Design

Cycle 4 is the reference fuel cycle for the nuclear and thermal-hydraulic analyses performed for Cycle 5 operation. There are no significant core design changes between Cycle 4 and Cycle 5. The only change is the increase in cycle lifetime to 460 EFPD. There are two significant operational changes: withdrawal of the ASPRs at 399 EFPD and a change from rodged to a feed-and-bleed mode of operation. These alter the core xenon stability. The results of an analysis of the stability and control of the core in the feed-and-bleed mode with ASPRs removed, shows the stability index is -0.0428h^{-1} . This demonstrates the axial stability of the core.

To support Cycle 5 operation of Crystal River Unit 3, the licensee has provided analyses (Ref. 2) using analytical techniques and design bases established in B&W reports that have been approved by the NRC staff. The licensee has provided a comparison of the core physics parameters for Cycles 4 and 5 as calculated with these techniques. We find the predicted characteristics acceptable because they use approved techniques, the validity of which has been reinforced through a number of cycles of predictions for this and other reactors. As a result of our review of the characteristics compared to previous cycles, we agree with their use in the Cycle 5 accident and transient analysis, as discussed in Section 5.

The predicted control rod worths differ between cycles due to changes in radial flux and burnup distributions. The licensee took into account ejected rod worths and their adherence to shutdown margin requirements in the development of rod position limits for Cycle 5. The maximum stuck rod worth for Cycle 5 is greater than that for design Cycle 4 at BOC and less at EOC. The licensee presented an analysis of shutdown margin adequacy as a function of predicted control and stuck rod worths. This analysis allowed for a 10 percent uncertainty on net rod worth and for flux redistribution. It shows considerable margin in excess of requirements.

We, therefore, conclude that the licensee has demonstrated adequate provision of shutdown margin for Cycle 5. In addition, control rod worth measurements are made during startup tests. These confirm the adequacy of predicted control rod worths.

4.0 Evaluation of Thermal-Hydraulic Design

The objective of the thermal-hydraulic review is to confirm that the design of the reload core has been accomplished using acceptable methods, and that acceptable safety margin is available from conditions which would lead to fuel damage during normal operation and anticipated transients.

The thermal-hydraulic models and methodology used for Cycle 5 are the same as used for Cycle 4. The rod bow Departure from Nucleate Boiling Ratio (DNBR) penalty was calculated using the interim rod bow penalty evaluation procedure approved in Reference 30. The burnup used to calculate the penalty was the highest assembly burnup in Cycle 5 of 20,464 MWd/MtU.

The important thermal-hydraulic parameters are the same for both Cycles 4 and 5 as summarized in Table 1. Based on the similarities of Cycles 4 and 5, we find the operation of Cycle 5 acceptable.

5.0 Technical Specifications Related to Cycle 5 Reload

As indicated in our review of Sections 3.0 and 4.0 above, the operating characteristics for Cycle 5 were calculated with well-established, approved methods. In addition, we agreed in Section 3 with the licensee's evaluation of control rod worths and their role in the establishment of control rod position limits. Most of the TS changes proposed in Reference 1, Attachment A and Reference 2 are a reflection of these analyses, and are, therefore, acceptable.

Table 1 Thermal-Hydraulic Design Conditions

	Cycle 4, 2544 Mwt	Cycle 5, 2544 Mwt
Design power level, Mwt ^(a)	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	106.5	106.5
Reference design radial x local power peaking factor, $F_{\Delta H}$	1.71	1.71
Reference design axial flow shape	1.5 cosine	1.5 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Densified active length, in. (a)	140.2	140.2
Average heat flux at 100% power, Btu/h-ft ²	176 x 10 ³	176 x 10 ³
Maximum heat flux at 100% power, Btu/h-ft ²	452 x 10 ³	452 x 10 ³
CHF correlation	BAW-2	BAW-2
Minimum DNBR, % power	2.05(112)	2.05(112)

(a) Used in analysis.

There are also several proposed TS changes based on reinstallation of the reactor coolant pump power monitors (RCPPM) trip. This trip was approved for Cycle 4 operation but was removed because the setpoints led to numerous spurious trips. As indicated in Section 6.0, we accept the licensee's analysis of this trip with a pump monitor delay time of 1.5 seconds. The proposed TS changes implementing this trip are, therefore, acceptable. Thus all of the TSs proposed in BAW-1767 are acceptable.

A correction was required to Table 3.3-2, Item No. 8, in the originally proposed TS submitted by the licensee. The response time for pump status based on RCPPM trip should have been 1.44 seconds instead of 1.5 seconds. This is the 1.5 second total response time minus the 60 milliseconds for release of the control rod drive roller nut from the lead screw. This correction was reported in Reference 29.

6.0 Evaluation of Accident and Transient Analysis

The licensee has examined each FSAR accident analysis with respect to changes in Cycle 5 parameters to determine their effect on the plant thermal performance during hypothetical transients. The key parameters having the greatest effect on determining the outcome of a transient or accident are the core thermal parameters, thermal-hydraulic parameters, and physics and kinetics parameters. Core thermal properties used in the FSAR accident analysis were design operating values based on calculational values plus uncertainties. Table 1 compares the thermal-hydraulic parameters for Cycles 4 and 5. These parameters are the same for both cycles. A comparison of the key kinetics parameters from the FSAR and Cycle 5 is provided in Table 7-1 of Reference 2. These comparisons indicate no significant changes or changes in the conservative direction, except for the initial conditions for the four-pump coastdown and locked-rotor accidents. We have reviewed an analysis of the four-pump coastdown analysis provided in Reference 29 and find it acceptable. The locked-rotor accident was reevaluated for Cycle 3 operation. This analysis remains valid for Cycle 5. The effects of fuel densification on the FSAR accident analysis have also been evaluated.

Generic LOCA analyses for the B&W 177-fuel assembly lowered-loop NSSS have been performed using the final acceptance criteria emergency core cooling system (ECCS) evaluation model (Ref. 31). These analyses used the limiting values of key parameters for all plants in the 177-FA lowered loop category and, therefore, are bounding for Crystal River Unit 3 Cycle 5 operation. Further details on plant-specific aspects of these analyses are discussed in Section 2.0.

A comparison of the radiological doses calculated for Cycle 5 to those previously reported for Cycle 3 shows that all Cycle 5 dose values are either bounded by the Cycle 3 values or are a small fraction of the 10 CFR 100 limits, i.e., below 30 REM to the thyroid and 2.5 REM to the whole body.

7.0 Conclusion - Core Reload

We conclude from the examination of Cycle 5 core thermal and kinetic properties, with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the Crystal River Plant's ability to operate safely during Cycle 5.

8.0 Evaluation of RCPPM Trip Time Response Testing Requirements

Earlier sections of this report assess increasing the design time delay of the RCPPM trip from 0.56 seconds to 1.44 seconds. Our Safety Evaluation issued with Amendment No. 55 to License No. DPR-72 (Ref. 32) revised the required response time of this trip from previous values and imposed a number of specific requirements for verifying response times by actual testing. This testing was to include the current and voltage sensor elements and the watt transducer and was to have been performed during each refueling outage. On the basis of a number of technical problems associated with testing the sensor elements, including potential destruction of pump seals, and in consideration of the significant increase in the design time response from 0.56 to 1.44 seconds, the licensee requested elimination of the time response testing requirements of the sensors and watt transducer (Ref. 29). In evaluating this request, we required the licensee to provide additional information regarding testing that will have been accomplished prior to Cycle 5 startup and bases for assigning design time delays for those components which would not be tested. The licensee has responded with the requested information (Ref. 33) and we have completed our evaluation of this request.

Prior to startup, the licensee has committed to test all RCPPM strings starting with the adjustable time delay relay through the Control Rod Drive (CRD) circuit breakers. The total time delay associated with this portion of the circuitry is 1305 milliseconds (ms). The adjustable time delay relays have been bench tested five times each with a nominal setting of 840 ms (not to exceed 890 ms) to assure repeatability. The adjustable range of these relays is 0.1 to 3 seconds. The assigned time response of the current and voltage sensors (which will not be tested) is 20 ms based on information supplied by the manufacturer of the devices. The time delay assigned to the watt transducer and bistable device is 115 ms and, as discussed earlier in this report, the time assumed for the CRD roller nuts is 60 ms. On the basis of the time delay relay bench test verifications, confidence that setting drift has not been a problem with the type of relays used, the conservatisms used in calculating and testing the cumulative time delays, we have concluded that the need for actual testing of the sensor devices and watt transducer is no longer applicable and is hereby deleted as a requirement on the Crystal River Unit 3 RCPPM trip circuits.

Table 3.3-2 of the Crystal River TS has been appropriately amended with a footnote to exclude time response testing of the sensors and watt transducer.

9.0 Assessment of Two Year Cycle Impact

The licensee, in Attachment D to Reference 1, requested a number of changes to the TSs which would change the frequency of various surveillance requirements from 18 months to 24 months. The reason given for the requested changes was to delete the necessity to perform mid-cycle shutdowns to perform 18-month surveillances in view of the longer projected lifetime of the Cycle 5 core. We have evaluated this request and denied it (Ref. 35) on the basis of inadequate justification. Surveillance requirements are based upon equipment reliability and not the length of core lifetime. Therefore, the requested changes have not been included in this amendment.

10.0 Miscellaneous TS Changes

10.1 Sodium Hydroxide (NaOH) Chemical Addition Verification

On April 13, 1982, the licensee informed the NRC staff that a periodic surveillance plan to determine the capability of the Spray Additive System to deliver the required flow to the Reactor Building Spray (RBS) System was being developed. This TS change adds the surveillance requirements for performing this test. The licensee determined that the safest and most effective method for performing the drawdown test is to change BS additive tanks (i.e., switch the NaOH from BST-2 to BST-1). This switch will allow the licensee to perform a complete drawdown test without contaminating the decay heat and/or reactor coolant system (RCS), which could occur if BST-2 contained the NaOH. This tank switch will also prevent the occurrence of a moderator dilution event from the inadvertent addition of NaOH solution from BST-2 to the RCS. BST-1 (the old Sodium Thiosulfate Tank) empties only into the RBS system, making moderator dilution from BST-1 improbable. This tank switch, however, requires a change in the NaOH concentration to assure the spray pH is within the TS limits of 7.2 to 11.0. The applicable TS page has therefore been changed.

10.2 Shutdown Margin Change

With the previous alignment of the Decay Heat Removal (DHR) and RBS systems at Crystal River Unit 3, accidental opening of the Engineered Safeguard actuated valves in the sodium hydroxide (NaOH) lines during DHR system operation could allow NaOH solution to enter the RCS resulting in a reduction of shutdown margin. Due to this possibility of inadvertent boron dilution, the SHUTDOWN MARGIN was restricted to greater than or equal to 3.5% delta k/k in MODES 4 and 5. The 3.5% delta k/k shutdown margin requirement, for MODES 4 and 5, assured that the reactor would not become critical if a flow path from NaOH tank BST-2 to the DHR system was established.

The possibility of a moderator dilution event by injection of NaOH solution into the DHR system has been greatly reduced by switching the NaOH solution from BST-2 to BST-1 (the old Sodium Thiosulfate Tank). NaOH, which initially gravity fed into the DHR and RBS system, now will only feed into the RBS system. Check valves in the RBS system prevent backflow into the DHR system. Thus, there is no need for the 3.5% delta k/k restriction and the shutdown margin requirement for MODES 4 and 5 can safely be changed to the same requirement as for MODES 1, 2 and 3 (1.0% delta k/k).

At the time that the possibility of a boron dilution event was discovered, the shutdown margins for MODES 4 and 5 were separated from those for MODES 1, 2, and 3. Because the shutdown margin requirements will again be consistent for MODES 1 through 5, the Specifications have been recombined.

Modifications to the plant (i.e., changing NaOH from BST-2 to BST-1) have made the boron dilution by injection of NaOH solution improbable. Now, instead of injecting NaOH solution into the DHR and RBS systems, the solution is injected only into the RBS system. The valve which isolates BST-2 from the RCS has been rendered inoperable and is under administrative controls.

Thus, due to plant modification and the insignificant safety risk of a boron dilution event, we have concluded that changing the SHUTDOWN MARGIN to greater than or equal to 1.0% delta k/k (consistent with MODES 1, 2 and 3) will not adversely affect plant safety.

10.3 Power Operated Relief Valve (PORV) Emergency Power Surveillance

Surveillance requirement 4.4.3.2.3 was inadvertently included in the Crystal River Unit 3 TS Amendment 55 (Ref. 32). The PORV and block valves do not have specific emergency power supplies. On a Loss of Offsite Power Event, the Emergency Diesel Generators automatically supply power to the safety-related buses powering the PORV and block valves. This change is only editorial.

10.4 Motor Operated Emergency Feedwater Pump Inclusion and Flow Path Verification

The surveillance requirements for the motor driven Emergency Feedwater pump are being added to the TSs at the request of the NRC staff (Ref. 36). The additional requirements of flow path verification are also made at the request of the NRC. These changes will not degrade plant safety. The additional surveillance requirements are already being performed. This change is administrative in nature.

10.5 Hydraulic Snubber Inspection Schedule

The failure rate associated with snubber inspection and testing over the last operating cycle was high enough to require a 124 day visual inspection interval following the 1983 refueling outage. However, snubber design modifications and maintenance changes were implemented during the 1983 refueling outage to eliminate the causes of the failures encountered during the previous operating cycle. The TS change will extend the required inspection interval from 124 days to between 4 and 10 months. This extension is justified by the design modifications and maintenance changes implemented during the 1983 refueling outage.

We conclude that plant safety will not be compromised by this TS change. The maintenance performed and the design changes implemented on the Crystal River Unit 3 hydraulic snubbers over the last two refueling outages provide a high assurance of snubber operability. The maintenance and design changes have also provided essentially "new" snubbers.

11.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

12.0 Conclusion

We have concluded, based on the considerations discussed above, that:
(1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 12, 1983

The following NRC personnel have contributed to this Safety Evaluation:
John Voglewede, Marvin Dunenfeld, Tai Huang, Tom Dunning and Ronald Hernan.

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