July 16, 1985

Docket No. 50-302

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Mr. Walter S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
& Fuel Management
Post Office Box 14042; M.A.C. H-2
St. Petersburg, Florida 33733

Dear Mr. Wilgus:

The Commission has issued the enclosed Amendment No. 77 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 25, 1985.

This amendment revises the TSs to support the operation of CR-3 at full rated power during Cycle 6 operation.

A copy of our Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Original signed by

Harley Silver, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No. 77 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. W. S. Wilgus
Florida Power Corporation

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Generating Plant

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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
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
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 April 25, 1985, 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.

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. 77 , 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 16, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 77

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.

Page

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2-7
B 2-1
B 2-2
B 2-5
B 2-6
B 2-8
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3/4 1-16
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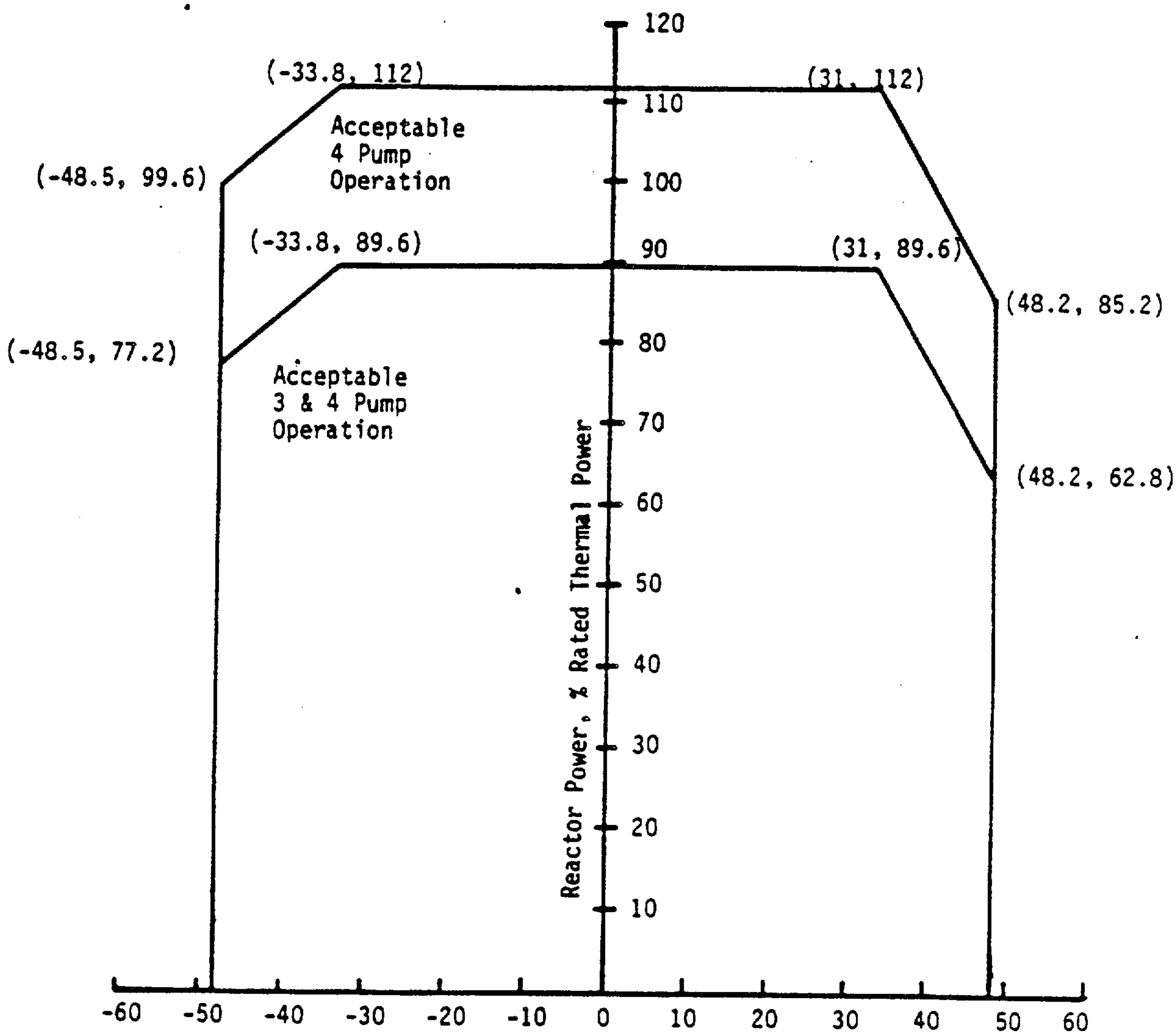
B 3/4 2-1
B 3/4 2-2
B 3/4 2-3
B 3/4 7-1
5-4

Figure 2.1-2
REACTOR CORE SAFETY LIMIT



 Amendments Nos. ~~10~~, ~~19~~, ~~32~~, ~~41~~, ~~48~~, ~~58~~, ~~64~~, 77

SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

2.2.1 The Reactor Protection System Instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

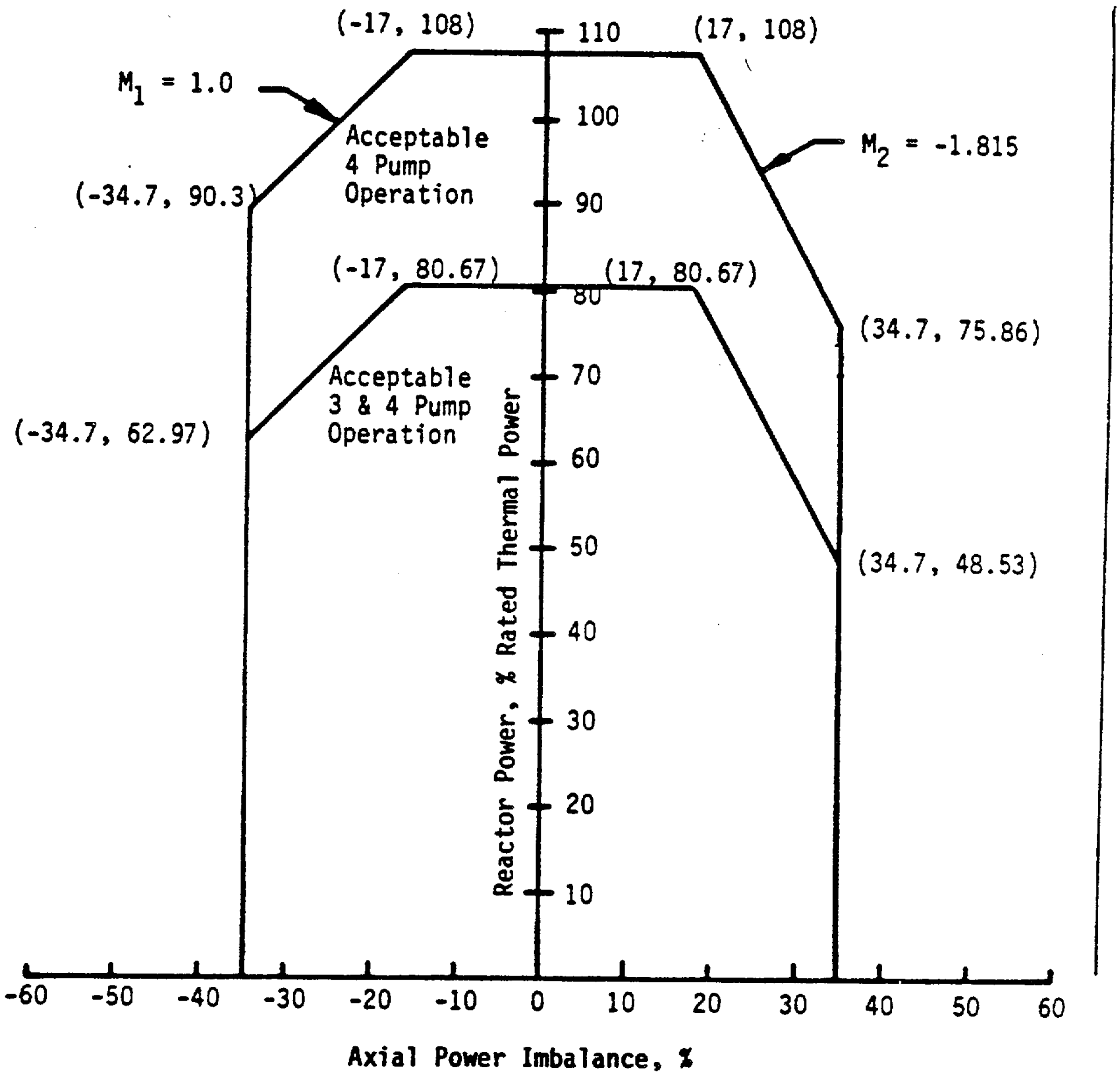
APPLICABILITY: As shown for each channel in Table 3.3-1.

ACTION:

With a Reactor Protection System instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

FIGURE 2.2-1

TRIP SETPOINT FOR NUCLEAR OVERPOWER BASED ON RCS FLOW AND AXIAL POWER IMBALANCE



2.1 SAFETY LIMITS

BASES

2.1.1 and 2.1.2 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime would result in excessive cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and, therefore, THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB through the BAW-2 DNB correlation. The DNB correlation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curve presented in Figure 2.1-1 represents the conditions at which a DNBR of 1.30 or greater is predicted for the maximum possible thermal power 112% when the reactor coolant flow is 139.7 x 10^6 lbs/hr, which is 106.5% of the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

$$F_Q^N = 2.82 \qquad F_{\Delta H}^N = 1.71 \qquad F_Z^N = 1.65$$

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

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

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

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of $F^N_Q = 2.82$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 20.5 kw/ft.

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

The specified flow rates for curves 1 and 2 of Figure 2.1-2 correspond to the expected minimum flow rates with four pumps and three pumps respectively.

The curve of Figure 2.1-1 is the most restrictive of all possible reactor coolant pump-maximum thermal power combinations shown in BASES Figure 2.1. The curves of BASES Figure 2.1 represent the conditions at which a minimum DNBR of 1.30 is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation.

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

The DNBR as calculated by the BAW-2 DNB correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher. Extrapolation of the correlation beyond its published quality range of 22% is justified on the basis of experimental data.

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip less than or equal to 618 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 108% and reactor flow rate is 100%, or flow rate is less than or equal to 95.88% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is greater than or equal to 80.67% and reactor flow rate is 74.7%, or flow rate is less than or equal to 69.44% and power is 75%.

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

BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.08% 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} ^{o}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} ^{o}F -5077.8) psig.

Reactor Containment Vessel Pressure - High

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

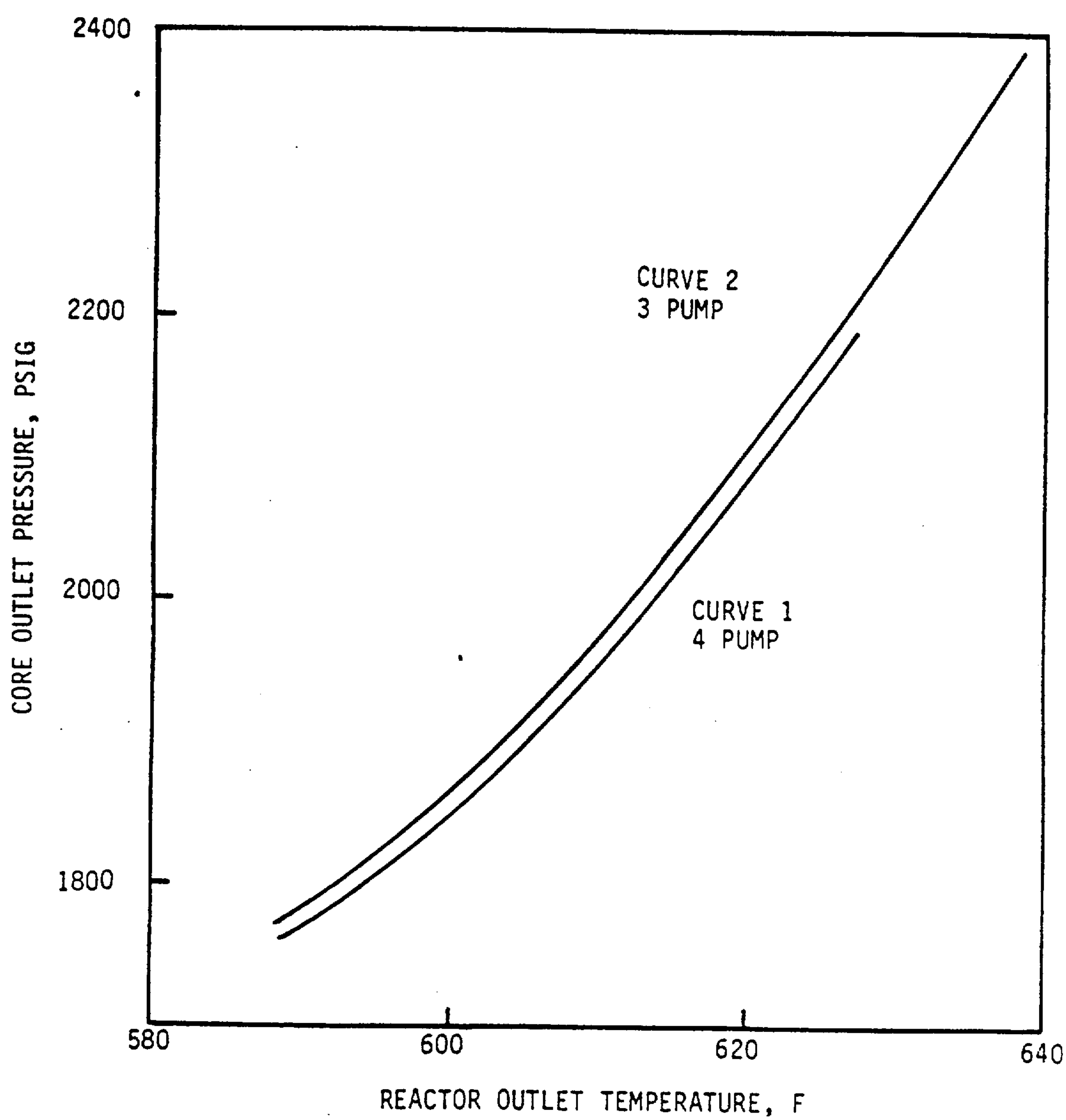
Reactor Coolant Pump Power Monitors

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.

BASES FIGURE 2.1

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM
ALLOWABLE POWER FOR MINIMUM DNBR



REACTOR COOLANT FLOW

CURVE	FLOW (% DESIGN)	POWER (RTP)	PUMPS OPERATING (TYPE OF LIMIT)
1	139.7×10^6 (106.5%)	112%	4 PUMPS (DNBR)
2	104.4×10^6 (79.6%)	89.6%	3 PUMPS (DNBR)

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 600 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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

BORATED WATER SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.9 Each of the following borated water sources shall be OPERABLE

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

1. A minimum contained borated water volume of 6,000 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.

ACTION: (Continued)

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

GROUP HEIGHT - AXIAL POWER SHAPING ROD GROUP

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

*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, 3.1-3, 3.1-3a, and 3.1-4, with a rod group overlap of 25 ± 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.

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 200 ± 10 EFPD

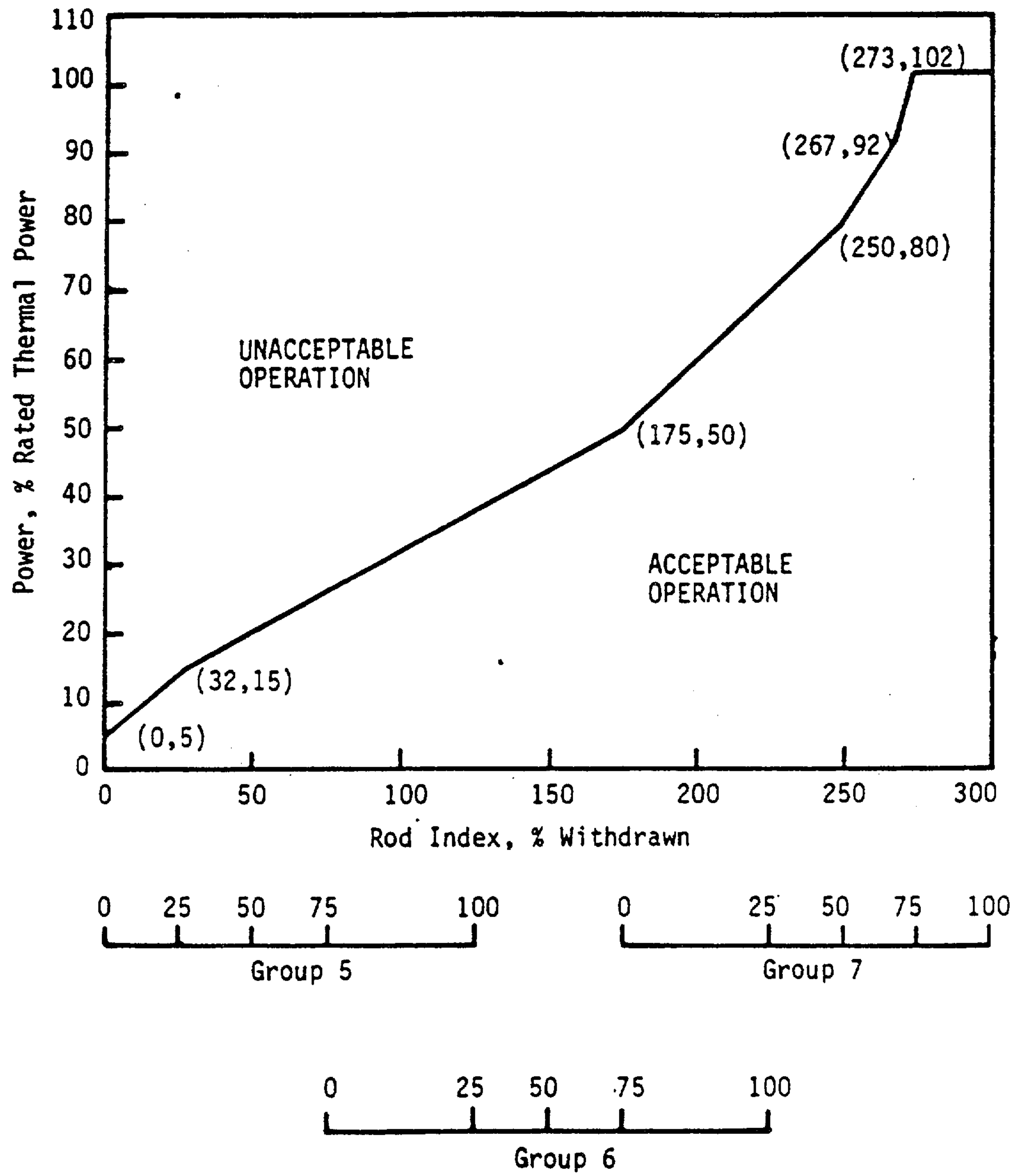


FIGURE 3.1-1a

REGULATING ROD GROUP INSERTION LIMITS FOR
FOUR PUMP OPERATION FROM 200 $\pm$10 TO 400 $\pm$10 EFPD

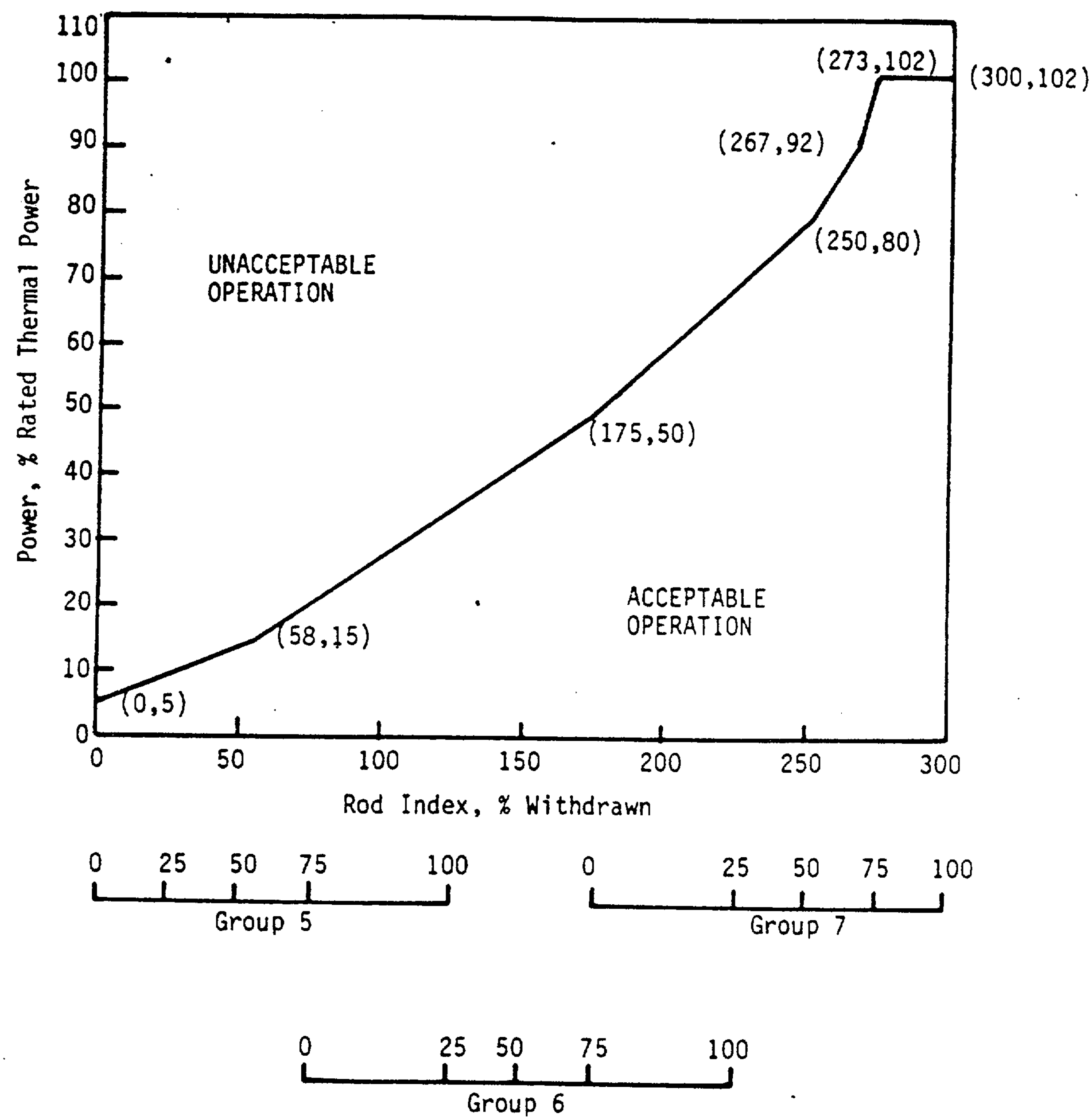


FIGURE 3.1-2

REGULATING ROD GROUP INSERTION LIMITS FOR
FOUR PUMP OPERATION AFTER 400 $\pm$10 EFPD

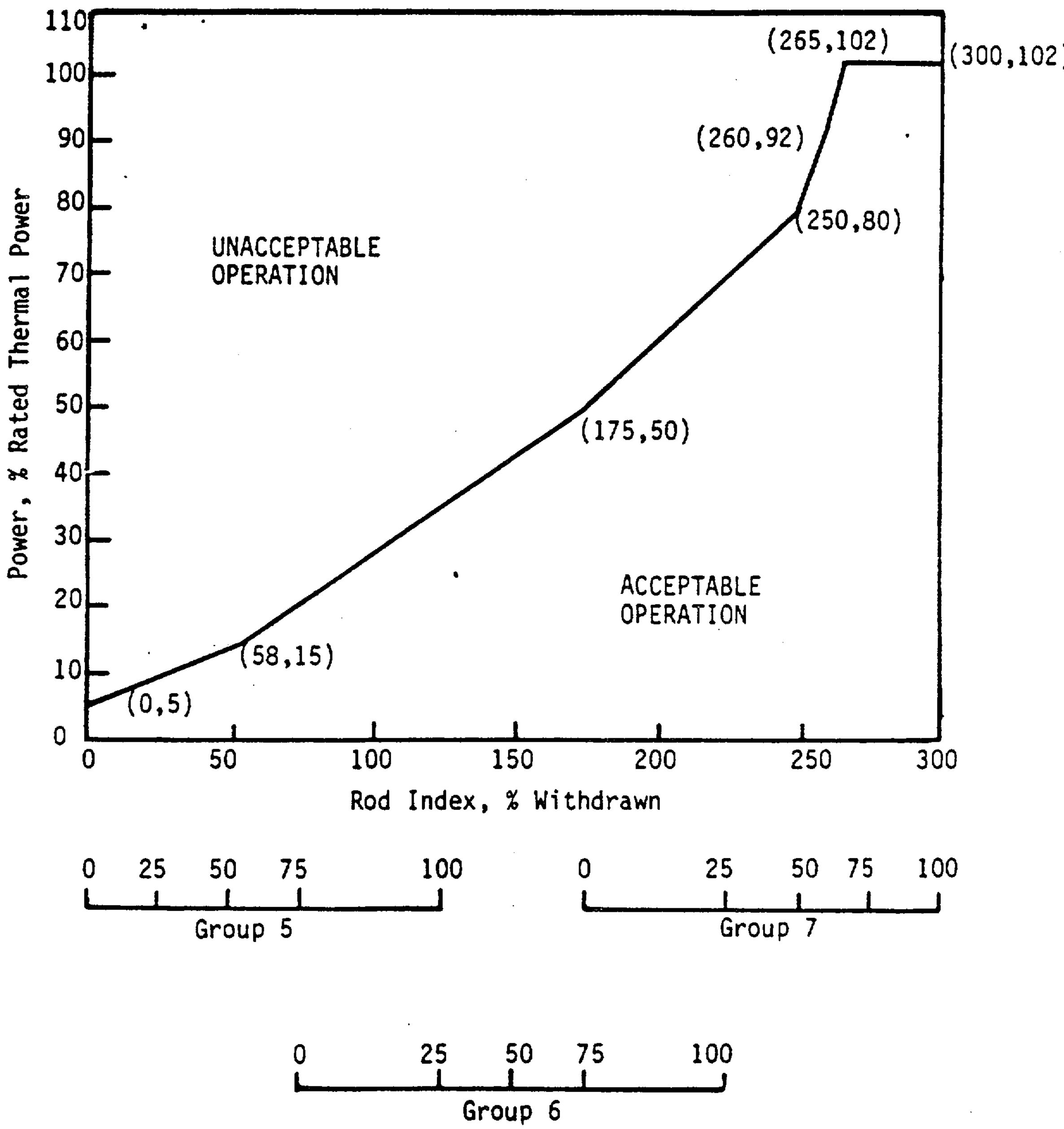


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FIGURE 3.1-3

REGULATING ROD GROUP INSERTION LIMITS FOR THREE PUMP OPERATION FROM 0 TO 200 $\pm$10 EFPD

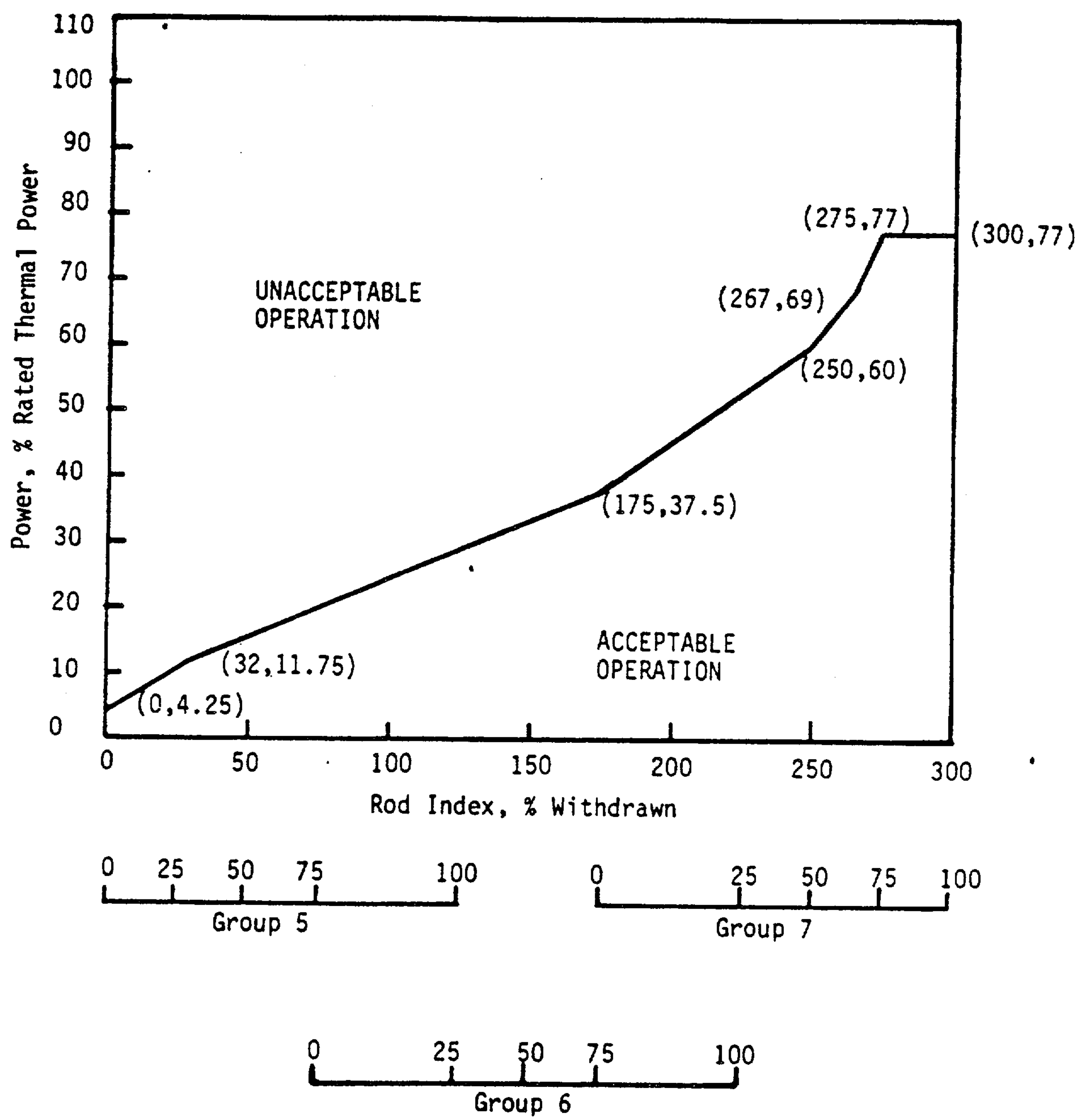


FIGURE 3.1-3a

REGULATING ROD GROUP INSERTION LIMITS FOR THREE PUMP OPERATION FROM 200 ±10 to 400 ±10 EFPD

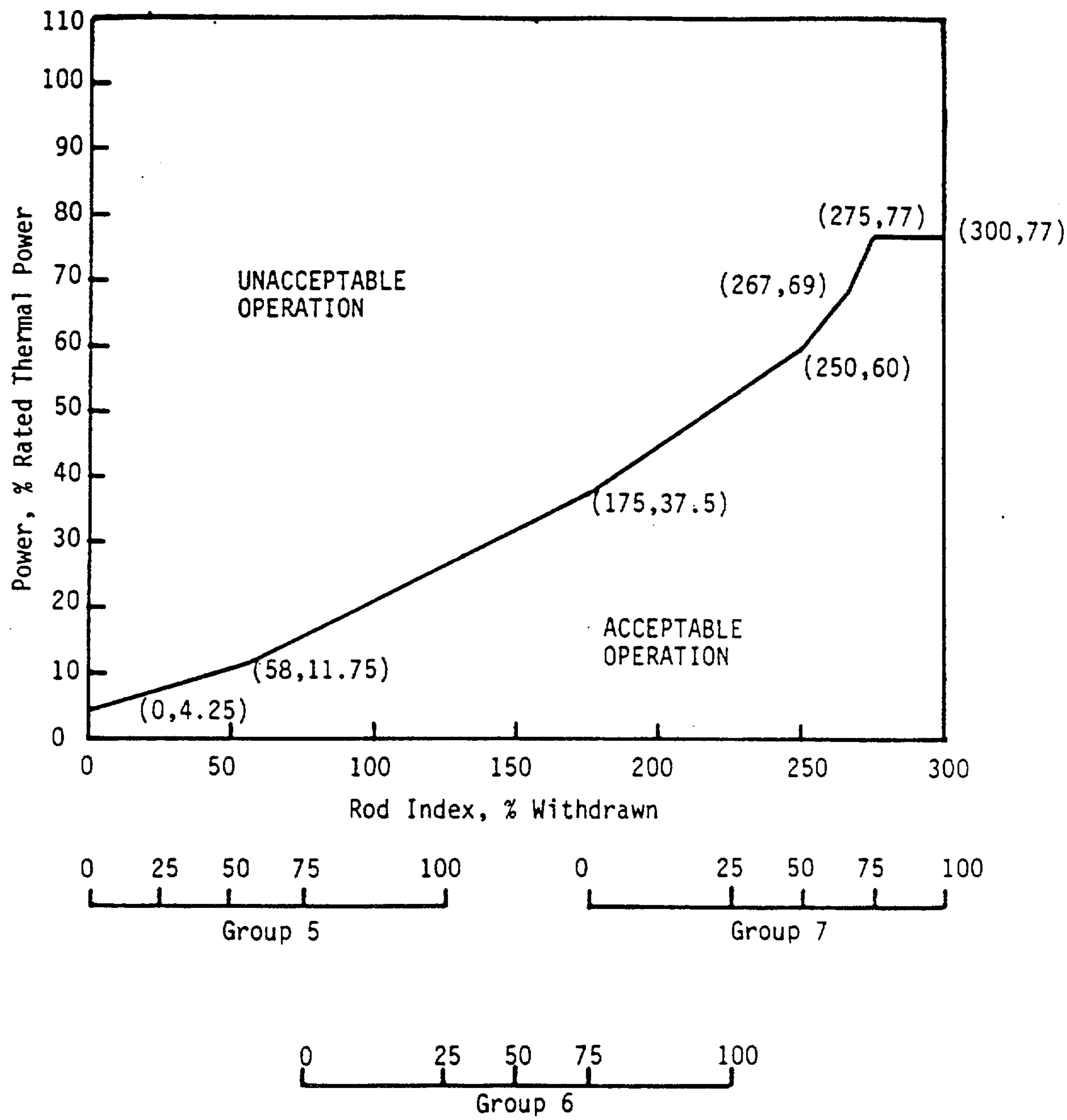
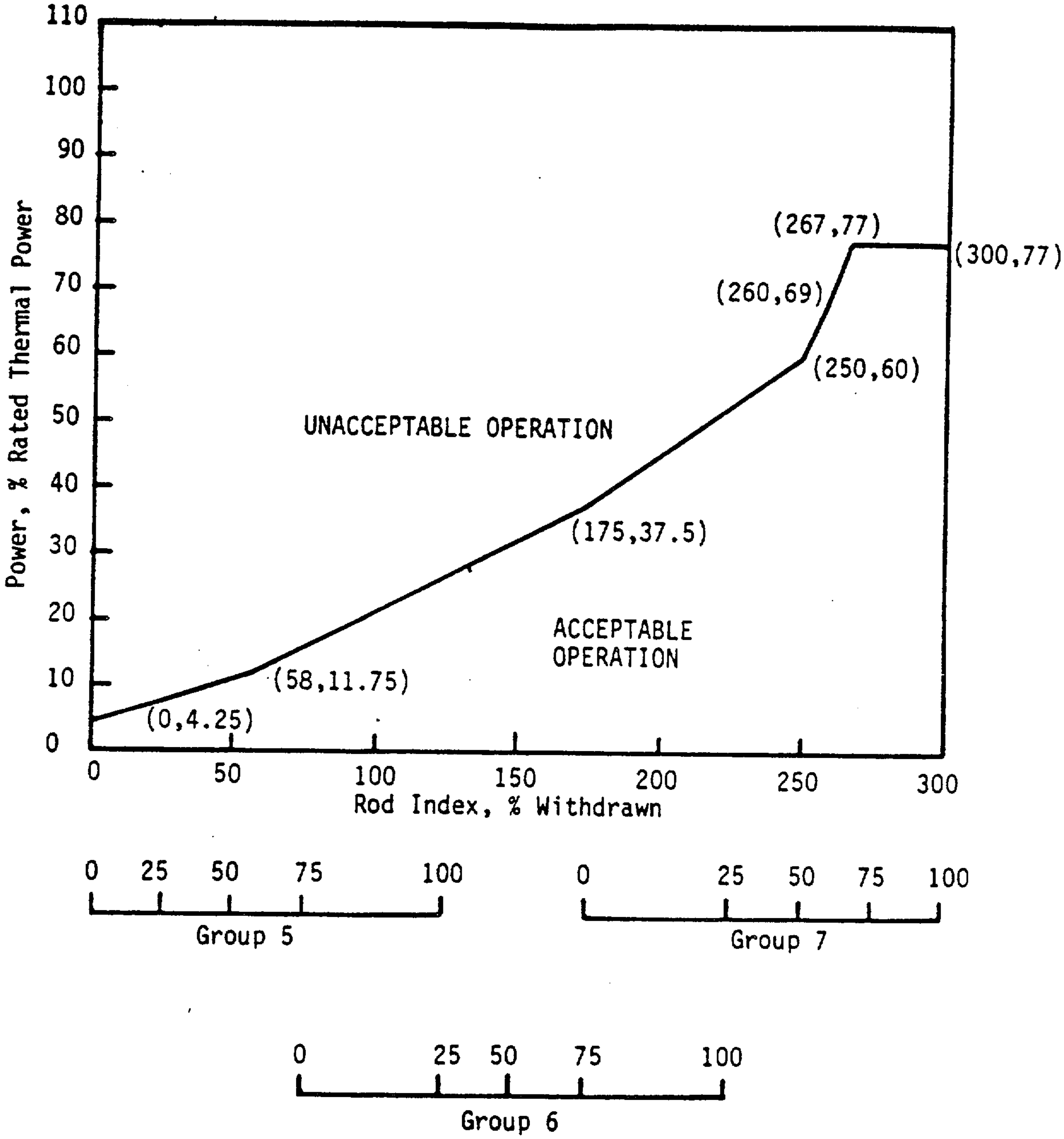


FIGURE 3.1-4

REGULATING ROD GROUP INSERTION LIMITS FOR
THREE PUMP OPERATION AFTER 400 ±10 EFPD



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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 patchs are locked subsequent to test, reprogramming or maintenance within the panels.

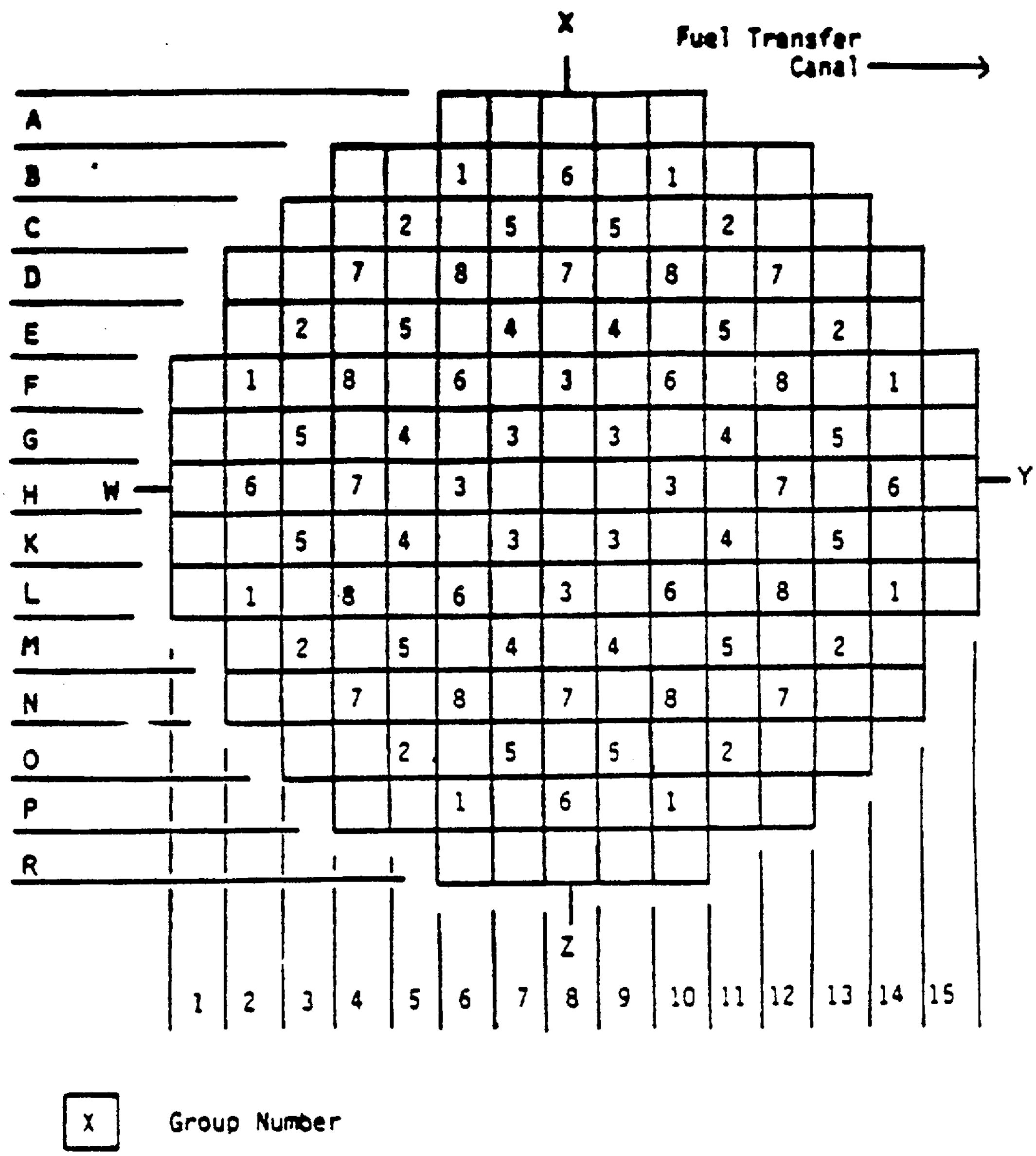
      b) For specifically affected individual rods, following maintenance, test, reconnection or modification of power or instrumentation cables from the control rod drive control system to the control rod drive.

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

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

*See Special Test Exceptions 3.10.1 and 3.10.2.

FIGURE 3.1-7
CONTROL ROD LOCATIONS AND GROUP
DESIGNATIONS FOR CRYSTAL RIVER 3 CYCLE 6



Group	No. of Rods	Function
1	8	Safety
2	8	Safety
3	8	Safety
4	8	Safety
5	12	Control
6	8	Control
7	8	Control
8	8	APSRs
	Total 68	

REACTIVITY CONTROL SYSTEMS

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3.9 Except as required for surveillance testing per Technical Specification 3.1.3.3, the following limits apply to axial power shaping rod (APSR) insertion. Up to 390 EFPD, the APSR's may be positioned as necessary. The APSR's shall be completely withdrawn (100%) by 410 EFPD. Between 390 and 410 EFPD, the APSR's may be withdrawn. However, once withdrawn during this period, the APSR's shall not be reinserted.

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 $K_{eff} \geq 1.0$.

-DELETED-

-DELETED-

-DELETED-

-DELETED-

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, 3.2-2, and 3.2-2a.

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 FOUR PUMP OPERATION FROM 0 TO 400 ±10 EFPD

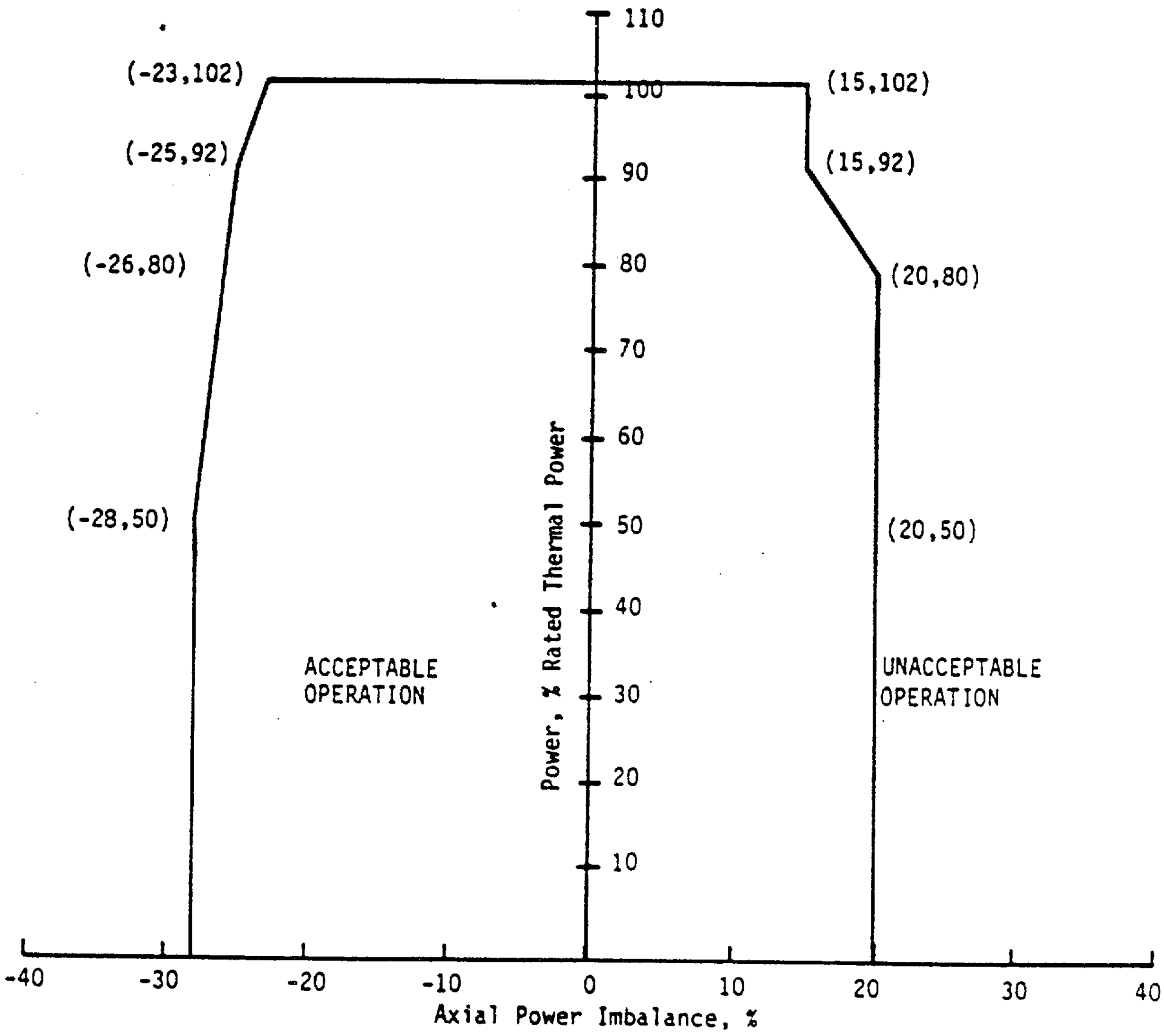


FIGURE 3.2-1a

AXIAL POWER IMBALANCE ENVELOPE FOR
FOUR-PUMP OPERATION AFTER 400 ±10 EFPD

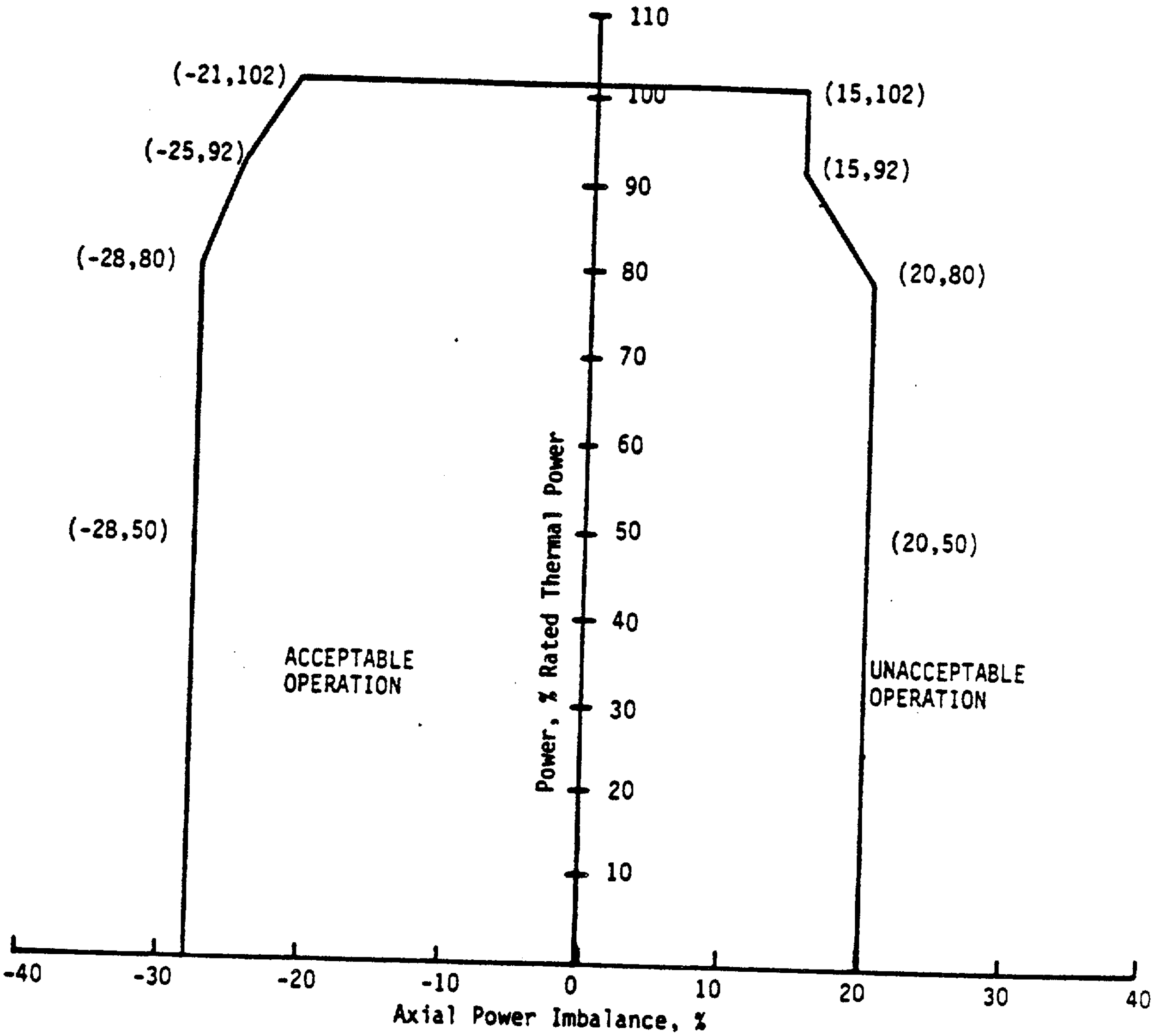


FIGURE 3.2-2

AXIAL POWER IMBALANCE ENVELOPE FOR
THREE PUMP OPERATION FROM 0 TO 400 ±10 EFPD

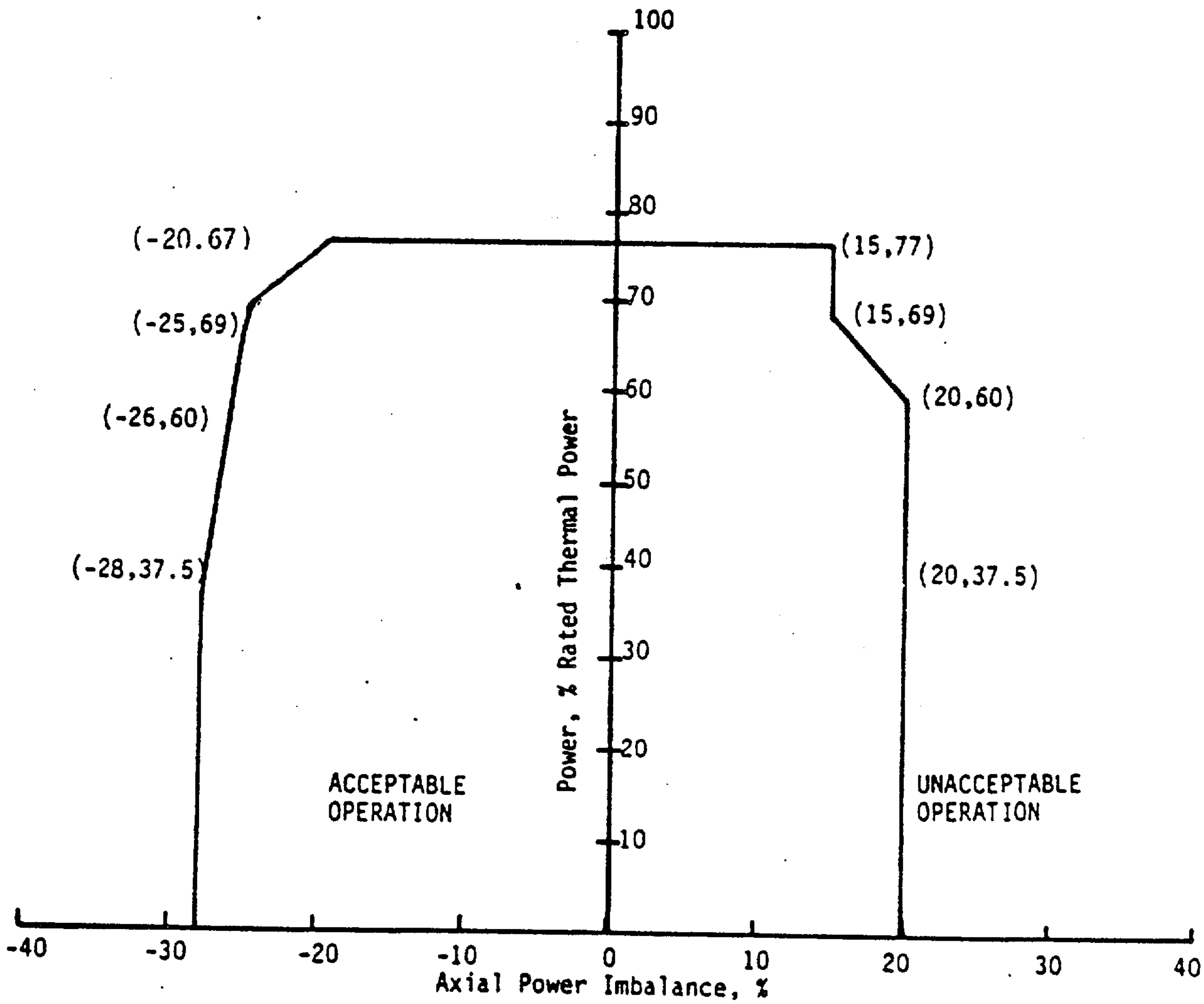
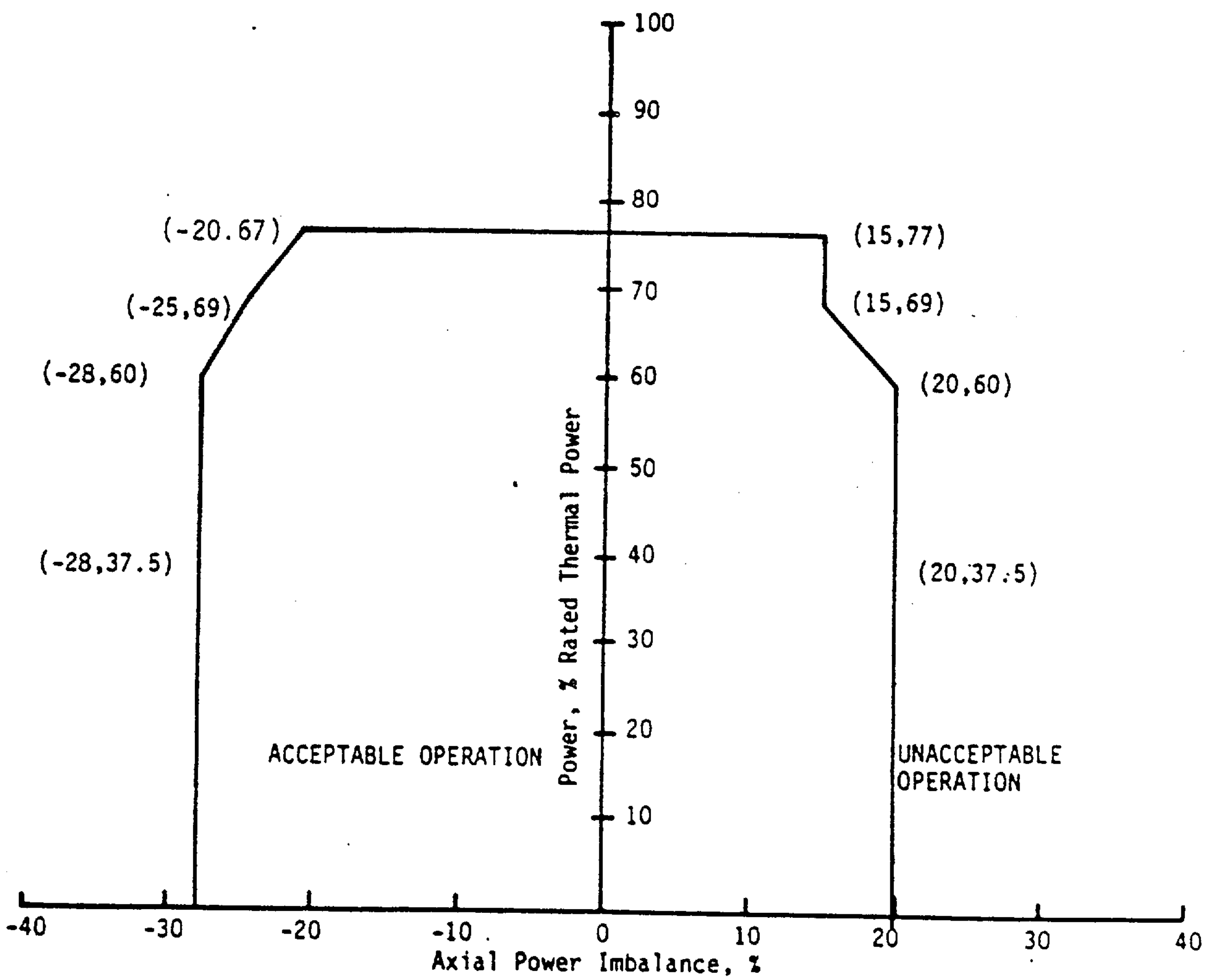


FIGURE 3.2-2a

AXIAL POWER IMBALANCE FOR THREE PUMP OPERATION AFTER 400 ±10 EFPD



POWER DISTRIBUTION LIMITS

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

$$F_Q \leq \frac{3.13}{P}$$

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

APPLICABILITY: MODE 1

ACTION:

With F_Q exceeding its limit:

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

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

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

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

c. The provisions of Specification 4.0.4 are not applicable.

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

POWER DISTRIBUTION LIMITS

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

LIMITING CONDITION FOR OPERATION

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

$$F^N_{\Delta H} \leq 1.71 \left[1 + 0.3\,(1-P)\right]$$

where $P = \dfrac{\text{Thermal Power}}{\text{Rated Thermal Power}}$

and $P \leq 1.0$

APPLICABILITY: MODE 1.

ACTION:

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

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

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

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

TABLE 3.2-2

QUADRANT POWER TILT LIMITS

	STEADY STATE LIMIT	TRANSIENT LIMIT	MAXIMUM LIMIT
QUADRANT POWER TILT as Measured by:			
Symmetrical Incore Detector System	3.20	9.08	20.0
Power Range Channels	1.61	6.96	20.0
Minimum Incore Detector System	1.73	4.40	20.0

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

a. Reactor Coolant Hot Leg Temperature

b. Reactor Coolant Pressure

c. Reactor Coolant Flow Rate

APPLICABILITY: MODE 1.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

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

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

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

ACTION 6 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, verify compliance with the SHUTDOWN MARGIN requirements of Specification 3.1.1.1 within one hour and at least once per 12 hours thereafter.

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

a. Within 1 hour:

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

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

b. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for 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*	$\leqq$0.266 seconds
3.	RCS Outlet Temperature - High	Not Applicable
4.	Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE*	$\leqq$1.842 seconds
5.	RCS Pressure - Low	$\leqq$0.44 seconds
6.	RCS Pressure - High	$\leqq$0.44 seconds
7.	Variable Low RCS Pressure	Not Applicable
8.	Pump Status Based on RCPPMs**	$\leqq$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 RCPPMs may exclude testing of the current and voltage sensors and the watt transducer.

TABLE 4.3-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

	FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
1.	Manual Reactor Trip	N.A.	N.A.	S/U(1)	N.A.
2.	Nuclear Overpower	S	D(2) and Q(7)	M	1, 2
3.	RCS Outlet Temperature--High	S	R	M	1, 2
4.	Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE	S(4)	M(3) and Q(7, 8)	M	1, 2
5.	RCS Pressure--Low	S	R	M	1, 2
6.	RCS Pressure--High	S	R	M	1, 2
7.	Variable Low RCS Pressure	S	R	M	1, 2
8.	Reactor Containment Pressure--High	S	R	M	1, 2
9.	Intermediate Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	1, 2 and *
10.	Source Range, Neutron Flux and Rate	S	R(7)	S/U(1)(5)	2, 3, 4 and 5
11.	Control Rod Drive Trip Breaker	N.A.	N.A.	M and S/U(1)	1, 2 and *
12.	Reactor Trip Module	N.A.	N.A.	M	1, 2, and *
13.	Shutdown Bypass RCS Pressure--High	S	R	M	2**, 3**, 4**, 5**
14.	Reactor Coolant Pump Power Monitors	S	R(9)	M	1, 2

TABLE 4.3-1 (Continued)

NOTATION

* - With any control rod drive trip breaker closed.

** - When Shutdown Bypass is actuated.

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

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

(3) - When THERMAL POWER (TP) is above 30% of RATED THERMAL POWER (RTP), compare out-of-core measured AXIAL POWER IMBALANCE (API_o) to incore measured AXIAL POWER IMBALANCE (API_I) as follows:

$$\frac{RTP}{TP} \quad (API_o - API_I) \quad = \text{Imbalance Error}$$

Recalibrate if the absolute value of the Imbalance Error is equal to or greater than 3.5%.

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

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

(6) - Each train tested every other month.

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

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

(9) - Current and voltage sensors may be excluded from CHANNEL CALIBRATION.

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
3. REACTOR BUILDING SPRAY				
a. Reactor Building Pressure High-High coincident with HPI Signal	S	R	M(4)	1,2,3
b. Automatic Actuation Logic	N/A	N/A	M(1) (3) (5)	1,2,3
4. OTHER SAFETY SYSTEMS				
a. Reactor Building Purge Exhaust Duct Isolation on High Radioactivity				
1. Gaseous	S	Q	M	All Modes
b. Steam Line Rupture Matrix				
1. Low SG Pressure	N/A	R	N/A	1,2,3
2. Automatic Actuation Logic	N/A	N/A	M (3)	1,2,3
c. Emergency Feedwater				
1. MFW Pump Turbines A and B Control Oil Low	S	R	N/A	1,2,3
3. OTSG A or B Level Low-Low	S	R	N/A	1,2,3,4

TABLE 4.3-2 (Cont'd)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
5. REACTOR BUILDING ISOLATION				
a. Manual Initiation	N/A	N/A	R	5 or 6
b. Reactor Building Pressure High	S	R	M(2)	1, 2, 3
c. Automatic Actuation Logic	N/A	N/A	M(1)(3)(5)	1, 2, 3, 4
d. Manual Initiation (HPI Isolation)	N/A	N/A	R	5 or 6
e. RCS Pressure Low (HPI Isolation)	S	R	M	1, 2, 3
f. Automatic Actuation Logic (HPI Isolation)	N/A	N/A	M(1)(3)(5)	1, 2, 3, 4

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

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

TABLE 4.3.6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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

TABLE 4.3-7
POST-ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Power Range Nuclear Flux	M	Q*
2. Reactor Building Pressure	M	R
3. Source Range Nuclear Flux	M	R*
4. Reactor Coolant Outlet Temperature	M	R
5. Reactor Coolant Total Flow Rate	M	R
6. RC Loop Pressure	M	R
7. Pressurizer Level	M	R
8. Steam Generator Outlet Pressure	M	R
9. Steam Generator Level (Primary EFW Flow Detector)	M	R
10. Borated Water Storage Tank Level	M	R
11. Startup Feedwater Flow Rate	M	R
12. Reactor Coolant System Subcooling Margin Monitor	M	R
13. PORV Position Indicator (Primary Detector)	M	R
14. PORV Position Indicator (Backup Detector)	M	R
15. PORV Block Valve Position Indicator	M	R
16. Safety Valve Position Indicator (Primary Detector)	M	R
17. Safety Valve Position Indicator (Backup Detector)	M	R
18. Emergency Feedwater Ultrasonic Flow Indicator (Backup EFW Fl ow Detector)	M	R

*Neutron detectors may be excluded from CHANNEL CALIBRATION

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.

or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

2. Degradation means a service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube.

3. Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

4. % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

5. Defect means an imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective. Any tube which does not permit the passage of the eddy-current inspection probe shall be deemed a defective tube.

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

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

8. Tube Inspection means an inspection of the entire steam generator tube as far as possible.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.4-2 (and Table 4.4-6, if the provisions of Specification 4.4.5.2.d are utilized).

4.4.5.5 Reports

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

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to Specification 6.9.2 within 12 months following the completion of the inspection. This Special Report shall include:

1. Number and extent of tubes inspected.

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

3. Identification of tubes plugged.

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

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

ECCS SUBSYSTEMS - $T_{avg} \geq 280°F$

LIMITING CONDITION FOR OPERATION

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

a. One OPERABLE high pressure injection (HPI) pump,

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

c. One OPERABLE decay heat cooler, and

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

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

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

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

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

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

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

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

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

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

   1. DHV-110
   2. DHV-111

e. At least once per 18 months by:

   1. Verifying automatic isolation and interlock action of the DHR system from the Reactor Coolant System when the Reactor Coolant System pressure is greater than or equal to 284 psig.

SURVEILLANCE REQUIREMENTS (Continued)

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

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

4. A visual inspection of the containment emergency sump which verifies that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

5. Verifying a total leak rate less than or equal to 6 gallons per hour for the LPI system at:

   a) Normal operating pressure or a hydrostatic test pressure of greater than or equal to 150 psig for those parts of the system downstream of the pump suction isolation valve, and

   b) Greater than or equal to 55 psig for the piping from the containment emergency sump isolation valve to the pump suction isolation valve.

f. At least once per 18 months during shutdown by

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

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

g. Following completion of HPI or LPI system modifications that could have altered system flow characteristics[1], by performance of a flow balance test during shutdown to confirm the following injection flow rates into the Reactor Coolant System:

HPI System - Single Pump

Single pump flow rate greater than or equal to 500 gpm at 600 psig.

While injecting through 4 Injection Legs, the flow rate for all combinations of 3 Injection Legs greater than or equal to 350 gpm at 600 psig.

LPI System - Single Pump

1. Injection Leg A - 2800 to 3100 gpm.

2. Injection Leg B - 2800 to 3100 gpm.

[1]Flow balance tests performed prior to complete installation of modifications are valid if performed with the system change that could alter flow characteristics in effect.

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

With one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided, that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Nuclear Overpower Trip Setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

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

TABLE 3.7-1
MAXIMUM ALLOWABLE NUCLEAR OVERPOWER TRIP SETPOINT WITH INOPERABLE STEAM LINE SAFETY VALVES

Maximum Number of Inoperable Safety Valves on Any Steam Generator	Maximum Allowable Nuclear Overpower Trip Setpoint (Percent of RATED THERMAL POWER)
1	96.35
2	81.95
3	67.5

PLANT SYSTEMS

3/4.7.9 HYDRAULIC SNUBBERS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With one or more hydraulic snubbers inoperable, replace or restore the inoperable snubber(s) to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

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

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

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

SURVEILLANCE REQUIREMENTS (Continued)

c. At least once per 18 months during shutdown a representative sample of at least 10 hydraulic snubbers or at least 10% of all snubbers listed in Table 3.7-3, whichever is less, shall be selected and functionally tested to verify correct piston movement, lock up and bleed. Snubbers greater than 50,000 lbs capacity may be excluded from functional testing requirements. Snubbers selected for functional testing shall be selected on a rotating basis. Snubbers identified in Table 3.7-3 as either "Especially Difficult to Remove" or in "High Radiation Zones" may be exempted from functional testing provided these snubbers were demonstrated OPERABLE during previous functional tests. Snubbers found inoperable during functional testing shall be restored to OPERABLE status prior to resuming operation. For each snubber found inoperable during these functional tests, an additional minimum of 10% of all snubbers or 10 snubbers, whichever is less, shall also be functionally tested until no more failures are found or all snubbers have been functionally tested.

TABLE 4.7-4

HYDRAULIC SNUBBER INSPECTION SCHEDULE

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

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

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

3/4.7.10 SEALED SOURCE CONTAMINATION

LIMITING CONDITION FOR OPERATION

3.7.10.1 Each sealed source containing radioactive material either in excess of 100 microcuries of beta and/or gamma emitting material or 5 microcuries of alpha emitting material shall be free of $\geq$ 0.005 microcuries of removable contamination.

APPLICABILITY: At all times.

ACTION:

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

   1. Either decontaminated and repaired, or

   2. Disposed of in accordance with Commission Regulations.

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

SURVEILLANCE REQUIREMENTS

4.7.10.1.1 Test Requirements - Each sealed source shall be tested for leakage and/or contamination by:

a. The licensee, or

b. Other persons specifically authorized by the Commission or an Agreement State.

The test method shall have a detection sensitivity of at least 0.005 microcuries per test sample.

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

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

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.

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 RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include (1) borated water sources, (2) makeup or DHR pumps, (3) separate flow paths, (4) boric acid pumps, (5) associated heat tracing systems, and (6) an emergency power supply from OPERABLE emergency busses.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective-action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% 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 4,980 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 35,681 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. Also the 6,000 gallons minimum BAST requirement per Specification 3.1.2.9 is conservative for this cycle.

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 390 gallons of 11,600 ppm boron from the boric acid storage system or 1,990 gallons of 2,270 ppm boron from the borated water storage tank. To envelop future cycle BWST and BAST contained borated water volume requirements, a minimum volume of 13,500 gallons and 600 gallons, respectively are specified.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and during short term transients, (b) maintaining the peak linear power density $\leq$ 18.0 kW/ft during normal operation, and (c) maintaining the peak power density $\leq$ 20.5 kW/ft during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

The power-imbalance envelope defined in Figures 3.2-1, 3.2-1a, 3.2-2, and 3.2-2a and the insertion limit curves, Figures 3.1-1, 3.1-1a, 3.1-2, 3.1-3, 3.1-3a, 3.1-4, 3.1-9, and 3.1-10 are based on LOCA analyses which have defined the maximum linear heat rate such that the maximum clad temperature will not exceed the Final Acceptance Criteria of 2200°F following a LOCA. Operation outside of the power-imbalance envelope alone does not constitute a situation that would cause the Final Acceptance Criteria to be exceeded should a LOCA occur. The power-imbalance envelope represents the boundary of operation limited by the Final Acceptance Criteria only if the control rods are at the insertion limits, as defined by Figures 3.1-1, 3.1-1a, 3.1-2, 3.1-3, 3.1-3a, 3.1-4, 3.1-9, and 3.1-10, and if the steady state limit QUADRANT POWER TILT exists. Additional conservatism is introduced by application of:

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

The conservative application of the above peaking augmentation factors compensates for the potential peaking penalty due to fuel rod blow.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The definitions of the design limit nuclear power peaking factors as used in these specifications are as follows:

F_Q Nuclear Heat Flux Hot Channel Factor, is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod on which minimum DNBR occurs to the average rod power.

It has been determined by extensive analysis of possible operating power shapes that the design limits on nuclear power peaking and on minimum DNBR at full power are met, provided:

$$F_Q \leq 3.13; \quad F^N_{\Delta H} \leq 1.71$$

Power Peaking is not a directly observable quantity and therefore limits have been established on the bases of the AXIAL POWER IMBALANCE produced by the power peaking. It has been determined that the above hot channel factor limits will be met provided the following conditions are maintained.

1. Control rods in a single group move together with no individual rod insertion differing by more than ± 6.5% (indicated position) from the group average height.

2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

3. The regulating rod insertion limits of Specification 3.1.3.6 and the axial power shaping rod insertion limits of Specification 3.1.3.9 are maintained.

4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained within the limits of Figures 3.2-1, 3.2-1a, 3.2-2, and 3.2.2a.

The design limit power peaking factors are the most restrictive calculated at full power for the range from all control rods fully withdrawn to minimum allowable control rod insertion and are the core DNBR design basis. Therefore, for operation at a fraction of RATED THERMAL POWER, the design limits are met. When using incore detectors to make power distribution maps to determine F_Q and $F^N_{\Delta H}$:

a. The measurement of total peaking factor, F_Q^{Meas}, shall be increased by 1.4 percent to account for manufacturing tolerances and further increased by 7.5 percent to account for measurement error.

b. The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$, shall be increased by 5 percent to account for measurement error.

For Condition II events, the core is protected from exceeding 20.5 kW/ft locally, and from going below a minimum DNBR of 1.30 by automatic protection on power, AXIAL POWER IMBALANCE, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the Reactor Protection System.

The QUADRANT POWER TILT limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation. For QUADRANT POWER TILT, the safety (measurement independent) limit for Steady State is 4.49, for Transient State is 11.07, and for the Maximum Limit is 20.0.

The QUADRANT POWER TILT limit at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. The limit was selected to provide an allowance for the uncertainty associated with the power tilt. In the event the tilt is not corrected, the margin for uncertainty on F_Q is reinstated by reducing the power by 2 percent for each percent of tilt in excess of the limit.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the FSAR initial assumptions and have been analytically demonstrated adequate to maintain a DNBR of 1.30 or greater throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

3/4.7.1 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:

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

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

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

A = maximum number of inoperable safety valves per steam generator.

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

X^1 = total required relieving capacity of each steam generator for 112% Rated Thermal Power in lbs/hour (6,160,000 lbs/hr).

NOTS = Nuclear Overpower Trip Setpoint specified in Table 2.2.1.

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

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The Reactor Containment building is designed and shall be maintained for a maximum internal pressure of 55 psig and a temperature of 281°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy - 4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2253 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.50 weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 60 safety and regulating and 8 axial power shaping (ASPR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing. The APSRs shall contain a nominal 63 inches of absorber material at their lower ends. The absorber material for the APSRs shall be 100% Inconel.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 77 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

In a letter dated April 25, 1985 (Ref. 1), Florida Power Corporation (the licensee) made application to modify the Technical Specifications for Crystal River Unit 3 to permit operation for a sixth cycle. The safety analyses performed and the resulting modifications to the plant Technical Specifications are described in the Cycle 6 reload report (Ref. 2).

The safety analysis for the previous fifth cycle of operation at Crystal River Unit 3 is being used by the licensee as a reference for the proposed sixth cycle of operation. Where conditions are identified as limiting in the fifth cycle analysis, our previous evaluation (Ref. 6) of that cycle continues to apply.

1.1 Description of the Cycle 6 Core

The Crystal River Unit 3 core consists of 177 fuel assemblies, each of which is a 15 x 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 6 will operate in a feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 60 full length control rod assemblies (CRAs) and 44 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.

1.2 Significant Areas of Review for this Reload

For the most part, Cycle 6 of Crystal River Unit 3 will be identical in operation to Cycle 5, and most Technical Specification changes such as reactor core safety limit, trip setpoints, rod insertion limits and imbalance limits are the result of the changes associated with the insertion of new fuel, cycle lifetime and the time of withdrawal of APSR(s) which are often made in Babcock & Wilcox (B&W) reactors. However, there were two significant changes associated with this reload. First, the APSRs were changed from using Ag-In-Cd to Inconel for neutron absorption. This change affects both the fuel design and nuclear performance of the core which are evaluated in Sections 2 and 3 of this SE. Second, the use of crossflow models which can predict flow redistribution effects in an open lattice reactor core are used to determine Departure from Nucleate Boiling Ratio (DNBR) margins. This change affects the thermal-hydraulic design of the core which is evaluated in Section 4 of this SE.

2.0 EVALUATION OF THE FUEL SYSTEM DESIGN

2.1 Fuel Assembly Mechanical Design

The 60 B&W Mark-B4 15 x 15 fuel assemblies loaded as Batch 7 are mechanically interchangeable with Batches 6A, 6B, 7A, and 7B 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. A comparison of the fuel design parameters for the various fuel assemblies in Cycles 5 and 6 can be found in Table 4-1 of Reference 2.

2.2 Fuel Rod Design

Although all batches in the Crystal River Unit 3 Cycle 6 core will utilize the same Mark-B4 fuel design and are mechanically interchangeable, the Batch 8 assemblies will incorporate a slightly higher average enrichment. The 60 assemblies will contain 3.49 w/o U-235. The cladding stress, strain and collapse analyses for the Cycle 6 fuel rod design are bounded by conditions previously analyzed for Crystal River Unit 3 using methods and limits previously reviewed and approved by the NRC. We find that no further review of these areas is necessary.

2.3 Fuel Thermal Design

All fuel in the Cycle 6 core is thermally similar. The fresh Batch 8 fuel inserted for Cycle 6 operation introduces no significant differences in fuel thermal performance relative to the fuel remaining in the core. The thermal analyses for all fuel were performed with the TACO2 code. Nominal undensified input parameters used in this methodology are provided in Table 4-1 (Ref. 2). Densification effects are accounted for in the TACO2 code densification model.

Linear heat rate (LHR) to fuel melt capability for all fuel was determined with the TACO2 fuel pin performance code. The analysis performed for Cycle 6 demonstrates that 20.5 kW/ft is a conservative limit to preclude centerline fuel melt (CFM) for all fuel batches.

The maximum fuel rod burnup at the end of cycle (EOC) 6 is predicted to be less than 40,500 MWd/mtu. Fuel rod internal pressure has been evaluated with TACO2 for the highest burnup fuel rod and is predicted to be less than the nominal reactor coolant system pressure of 2200 psia.

All fuel thermal design analyses have been performed using TACO2 which has been previously reviewed and approved by the NRC staff (Ref. 4) and is therefore acceptable.

2.4 Gray APSR Design

The gray APSRs that are to be used in Cycle 6 were designed to improve creep life. Cladding thickness and rod ovality control, which are the primary factors controlling the creep life of a stainless steel material, have been improved to extend the creep life of the gray APSR. The minimum design cladding thickness of the Mark-B APSR is 18 mils, while that of the gray APSR is 24 mils. Additionally, the gap width between the end plug and the Inconel absorber material was reduced. Finally, the ovality in the gap area will also be controlled to tighter tolerances. The gray APSR design was analyzed by B&W to demonstrate that it meets specified design requirements. The APSR was analyzed for cladding stress due to pressure, temperature, and ovality. It was found that the gray APSR has sufficient cladding and weld stress margins. The gray APSR was also analyzed for cladding strain due to thermal and irradiation swelling. The results of B&W analysis showed that no cladding strain is induced due to thermal expansion or irradiation swelling of the Inconel absorber.

We have reviewed the mechanical design of the gray APSRs as presented by B&W in Reference 2 and find it acceptable.

2.5 Operating Experience

B&W has accumulated operating experience with the Mark-B 15 x 15 fuel assembly at all of the eight operating B&W 177-fuel assembly plants. A summary of this operating experience as of October 31, 1984, is given on page 4-4 of Reference 2.

3.0 EVALUATION OF THE NUCLEAR DESIGN

Table 5-1 (Ref. 2) compares the core physics parameters for Cycles 5 and 6 designs. The cycle burnup (BOC to EOC) will be smaller for Cycle 6 than for Cycle 5 because of the shorter Cycle 6 length. Differences in cycle length, feed batch size and enrichment, BPRA loading, shuffle pattern, and rod group designations for Cycle 6 account for the differences in the physics parameters from those of Cycle 5. The critical boron concentrations for Cycles 5 and 6 are given in Table 5-1.

The control rod worths differ between cycles due to the gray APSRs and changes in radial flux and burnup distributions. Calculated ejected rod worths and their adherence to criteria are considered by B&W at all times in life and at all power levels in the development of the rod position limits presented in Section 8 of Reference 2. The maximum stuck rod worths are less at BOC 6 and greater at EOC 6 than those for Cycle 5. The adequacy of the shutdown margin with Cycle 6 stuck rod worths is demonstrated in Table 5-2 of Reference 2. The following conservatisms are applied to B&W's shutdown calculations:

1. Poison material depletion allowance.
2. Ten percent uncertainty on net rod worth.
3. Flux redistribution penalty.

Flux redistribution was accounted for since the shutdown analysis was calculated using a two-dimensional model. The shutdown calculation at the end of Cycle 6 was analyzed at 400 EFPD and EOC. The latest time (±10 EFPD) in core life at which the APSRs are inserted will be 400 EFPD.

To support Cycle 6 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 5 and 6 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 6 accident and transient analysis, as discussed in Section 6 of this Safety Evaluation.

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.

Except for the steady state analysis codes discussed below, the thermal-hydraulic models and methodology used for Cycle 6 are the same as used for Cycle 5. The effect of rod bow on DNBR was accounted for using the analysis presented in Reference 5 which has been reviewed and approved by the NRC staff.

The important thermal-hydraulic parameters are very similar with the exception of the design axial peak (1.50→1.65) for Cycles 5 and 6 as summarized in Table 1. However, the principal steady state thermal-hydraulic analysis codes were changed from CHATA, TEMP to LYNXT, LYNX 1, and LYNX 2. LYNX 1 and LYNX 2 have been previously reviewed and approved for this application by the NRC staff. The LYNXT computer code is currently under NRC staff review. However, this review has progressed sufficiently to approve its use in the steady state mode for Cycle 6 of Crystal River Unit 3. LYNXT was previously approved by the staff for the steady state analysis of Arkansas Unit 1 Cycle 7. The use of LYNXT in the transient mode is currently being questioned by the staff and LYNXT may not be approved for such application. This is not a problem for Crystal River Cycle 6 since the previously reviewed and approved code RADAR was used to perform the transient thermal-hydraulic analysis of the core.

The minimum DNBR at the design overpower condition is equal to 2.07. Although the design axial peak has been increased from 1.50→1.65, the benefits of crossflow analyses have resulted in additional DNBR margins relative to Cycle 5.

TABLE 1

CRYSTAL RIVER UNIT 3
THERMAL-HYDRAULIC DESIGN ANALYSIS

	CYCLE 5	CYCLE 6
RATED POWER, MWt	2544	2544
DESIGN POWER, MWt	2568	2568
REACTOR COOLANT FLOW, GPM	374880	374880
EFFECTIVE FLOW FOR HEAT TRANSFER, %	91.9	90.9
REFERENCE DESIGN FΔH	1.71	1.71
REFERENCE DESIGN AXIAL POWER SHAPE	1.50 COSINE	1.65 COSINE
CHF CORRELATION	B&W-2	B&W-2
DESIGN DNBR LIMIT	1.30	1.30
PRINCIPAL T-H ANALYSIS CODES	CHATA, TEMP	LYNXT, LYNX1, LYNX2
MINIMUM DNBR @ 112% OVER POWER	2.05	2.07
MINIMUM DNBR @ CORE PROTECTION SAFETY LIMITS	>1.4	1.6
LIMITING TRANSIENT DNBR	>1.7	1.9
TRANSIENT ANALYSIS CODE	RADAR	RADAR

Based on the similarities between the major thermal-hydraulic parameters of Cycles 5 and 6 and use of approved and/or acceptable (LYNXT) analysis methods, we find the thermal hydraulic performance of Cycle 5 to be acceptable.

5.0 TECHNICAL SPECIFICATIONS

As indicated in our review, the operating characteristics for Cycle 6 were calculated with well-established, approved and/or acceptable 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. The Technical Specification changes proposed in References 1 and 2 are a reflection of these analyses and are therefore acceptable.

The licensee's submittal proposes changes to support operating Cycle 6 for Crystal River Unit 3. These changes include:

1. Reactor core safety limits and trip setpoints for reactor thermal power and axial power imbalance.

2. Minimum boric acid and borated water volumes.

3. Regulating and axial power shaping rod group insertion limits.

4. Axial power imbalance limits.

6.0 EVALUATION OF ACCIDENT AND TRANSIENT ANALYSIS

The licensee has examined each FSAR accident analysis with respect to changes in Cycle 6 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 5 and 6. These parameters are the same for both cycles. A comparison of the key kinetics parameters from the FSAR and Cycle 6 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 coast-down and locked-rotor accidents. However, the B&W analysis in Reference 2 finds that the locked-rotor accident evaluated at 102% of 2568 MWt for Cycle 3 operation remains valid for Cycle 6. Also, the Cycle 4 four-pump coastdown analysis, performed with an initial power level of 102% of 2544 MWt and a pump monitor delay time of 1.5 seconds, bounds Cycle 6 and remains valid. In addition, the single reactor coolant pump coastdown analysis accounted for equipment errors and delay times associated with the Rosemont flow transmitters, which replace the BY transmitters used in previous cycles. These analyses have been previously reviewed and approved by the NRC staff and are therefore acceptable.

Generic LOCA analyses for the B&W 177-fuel assembly (FA) lowered-loop NSSS have been performed using the final acceptance criteria emergency core cooling systems (ECCS) evaluation model. 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 Cycle 6 operation. Further details on plant-specific aspects of these analyses are discussed in Section 7.2 of Reference 2.

A comparison of the radiological doses calculated for Cycle 6 to those previously reported in the FSAR shows that all Cycle 5 dose values are either bounded by FSAR values or are a small fraction (10%) of the 10 CFR 100 limits. Thus the radiological impact of accidents during Cycle 6 are not significantly different from those described in Chapter 14 of the FSAR.

7.0 EVALUATION FINDINGS

We conclude from the examination of Cycle 6 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 ability of Crystal River Unit No. 3 to operate safely during Cycle 6.

8.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

9.0 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 16, 1985

Principal contributors: G. Schwenk

REFERENCES

1. Letter from G. R. Westafer (Florida Power) to H. R. Denton (NRC), "Technical Specification Change Request No. 135," dated April 25, 1985.

2. "Crystal River Unit 3 Reload Report," B&W Company Report BAW-1860, dated April 1985.

3. J. F. Stolz (NRC) letter to J. A. Hancock (Florida Power) transmitting Amendment 48 to Facility Operating License No. DPR-72, dated December 4, 1981.

4. C. O. Thomas (NRC) letter to J. H. Taylor (Babcock & Wilcox) transmitting "Safety Evaluation of BAW-10141," dated April 13, 1983.

5. J. C. Maxler, et al., "Fuel Rod Bowing in B&W Fuel Designs, Revision 1," Babcock & Wilcox, Lynchburg, Virginia, BAW-10147-P-A.

6. J. F. Stolz (NRC) letter to W. S. Wilgus (FPC) transmitting Amendment No. 64 with Safety Evaluation of Crystal River Unit 3 Cycle 5 Reload, dated July 12, 1983.