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RECEIVED DEC 3 1981 U.S. NUCLEAR REGULATORY COMMISSION

Mr. J. A. Hancock
Assistant Vice President-Nuclear
Operations
Florida Power Corporation
P. O. Box 14042, M.A.C.H.2.
St. Petersburg, Florida 33733

Dear Mr. Hancock:

The Commission has issued the enclosed Amendment No. 46 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated November 16, 1981, as supplemented November 20, 1981. The amendment revises the TSs to authorize Cycle 4 operation of CR-3 following refueling. Copies of the Safety Evaluation and the Notice of Issuance for the amendment are also enclosed.

Also, as discussed with and agreed to by your staff, TS 4.6.2.2.d, surveillance requirement for containment spray additive system, has been deleted because it performs no meaningful function. The measurement of a flow rate through the small drain valves BSV-101 and BSV-102 does not verify the capability of the Spray Additive System to deliver the required amount of sodium hydroxide following a loss-of-coolant accident.

The adequacy of the Spray Additive System was previously evaluated and determined acceptable by Amendment No. 20 dated July 3, 1979. Verification of flow capability on a periodic basis, however, was not discussed. We, therefore, request that FPC provide within 90 days an alternative surveillance plan to verify the capability of the Containment Spray Additive System to deliver the required flow to the Containment Spray System.

cp
1

Sincerely,

"ORIGINAL SIGNED BY
JOHN F. STOLZ"

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

8112220105 811204
PDR ADOCK 05000302
P PDR

FR NOTICE
+ AMENDMENT
ONLY

Enclosures:
1. Amendment No. 46
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD-OR:DL	OELD 12/3/81
SURNAME	RIngram/cb	PErickson	JStolz	TNovak	M. Karman
DATE	12/ /81	12/2/81*	12/3/81	12/3/81	

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Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Peter B. Erickson, Project Manager
Operating Reactors Branch #4
Division of Licensing

Enclosures:
1. Amendment No.
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

OFFICE	ORB#4:DL	ORB#4:DL PBE	C-ORB#4:DL	AD-OR:DL	OELD		
SURNAME	RIngram	PErickson/cb	JStolz	TNovak			
DATE	11/ /81	11/21/81	11/ /81	11/ /81	11/ /81		

Crystal River Unit No. 3
Florida Power Corporation

50-302

cc w/enclosure(s):

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Vice President and General Counsel
P. O. Box 14042
St. Petersburg, Florida 33733

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
Iverness, Florida 36250

Regional Radiation Representative
EPA Region IV
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Crystal River Public Library
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Crystal River, Florida 32629

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
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Mr. Tom Stetka, Resident Inspector
U.S. Nuclear Regulatory Commission
Route #3, Box 717
Crystal River, Florida 32629

Mr. Dan C. Poole
Nuclear Plant Manager
Florida Power Corporation
P. O. Box 219
Crystal River, Florida 32629

cc w/enclosure(s) & incoming dtd.:
11/16/81 & 11/20/81

Bureau of Intergovernmental Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

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

December 4, 1981

Docket No. 50-302

Mr. J. A. Hancock
Assistant Vice President-Nuclear
Operations
Florida Power Corporation
P. O. Box 14042, M.A.C.H.2.
St. Petersburg, Florida 33733

Dear Mr. Hancock:

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The adequacy of the Spray Additive System was previously evaluated and determined acceptable by Amendment No. 20 dated July 3, 1979. Verification of flow capability on a periodic basis, however, was not discussed. We, therefore, request that FPC provide within 90 days an alternative surveillance plan to verify the capability of the Containment Spray Additive System to deliver the required flow to the Containment Spray System.

Sincerely,

John F. Stolz

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

Enclosures:
1. Amendment No. 46
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

ATTACHMENT TO LICENSE AMENDMENT NO. 46

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.

2-3
2-7
B2-6
3/4 1-2a
3/4 1-7
3/4 1-10a
3/4 1-13
3/4 1-14
3/4 1-16
3/4 1-17
3/4 1-25
3/4 1-27
3/4 1-27a (new page)
3/4 1-28
3/4 1-29
3/4 1-29a (new page)
3/4 1-30
3/4 1-37
3/4 1-38
3/4 1-38a (new page)
3/4 1-39
3/4 2-1
3/4 2-2
3/4 2-2a (new page)
3/4 2-3

3/4 3-6

3/4 4-1
3/4 6-13
B3/4 1-2
B3/4 1-3

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. 46
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 November 16, 1981, as supplemented November 20, 1981, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

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P PDR

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

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 46 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: December 4, 1981

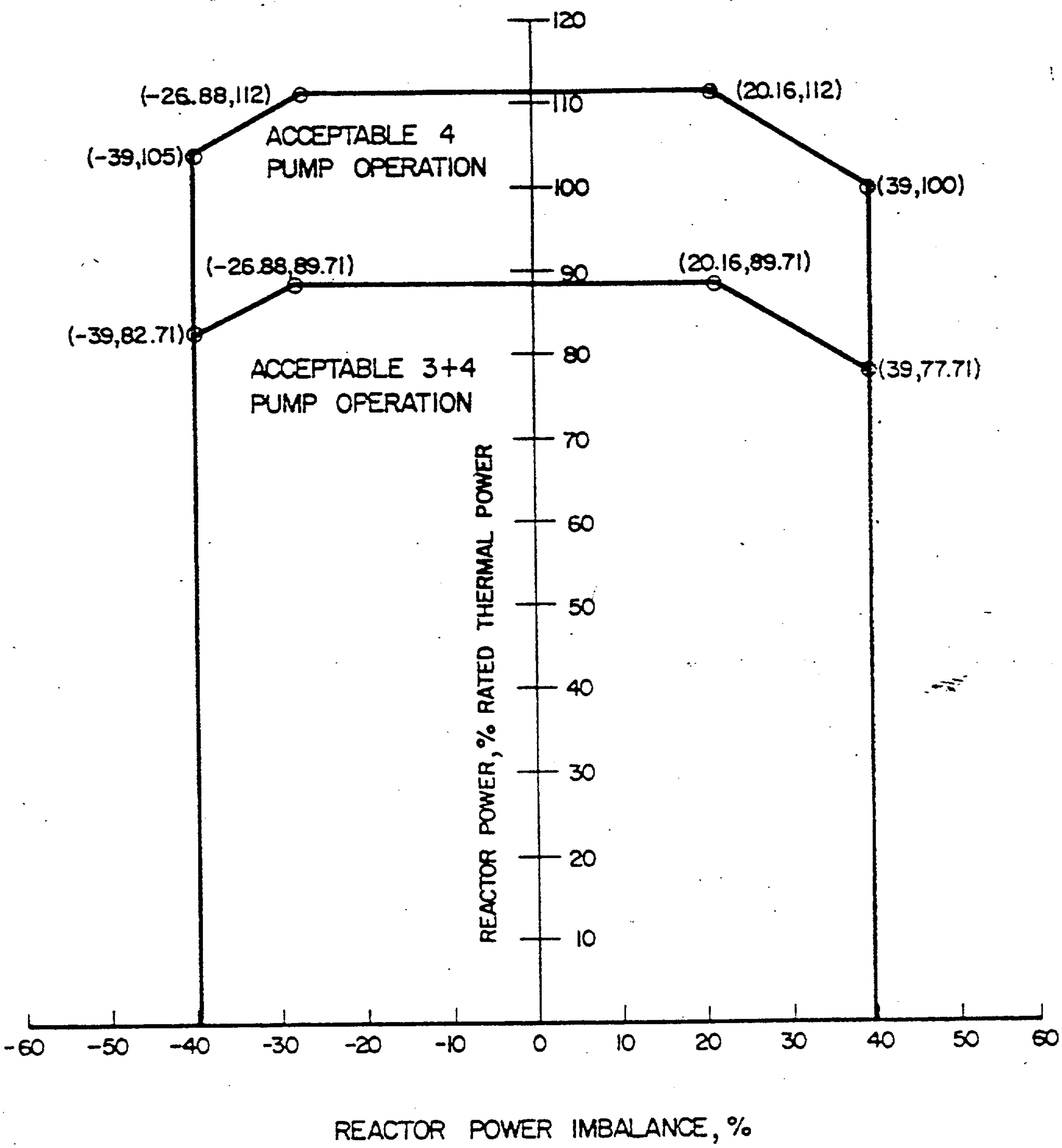

FIGURE 2.1-2

REACTOR CORE SAFETY LIMIT

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM SETPOINTS

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

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

ACTION:

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

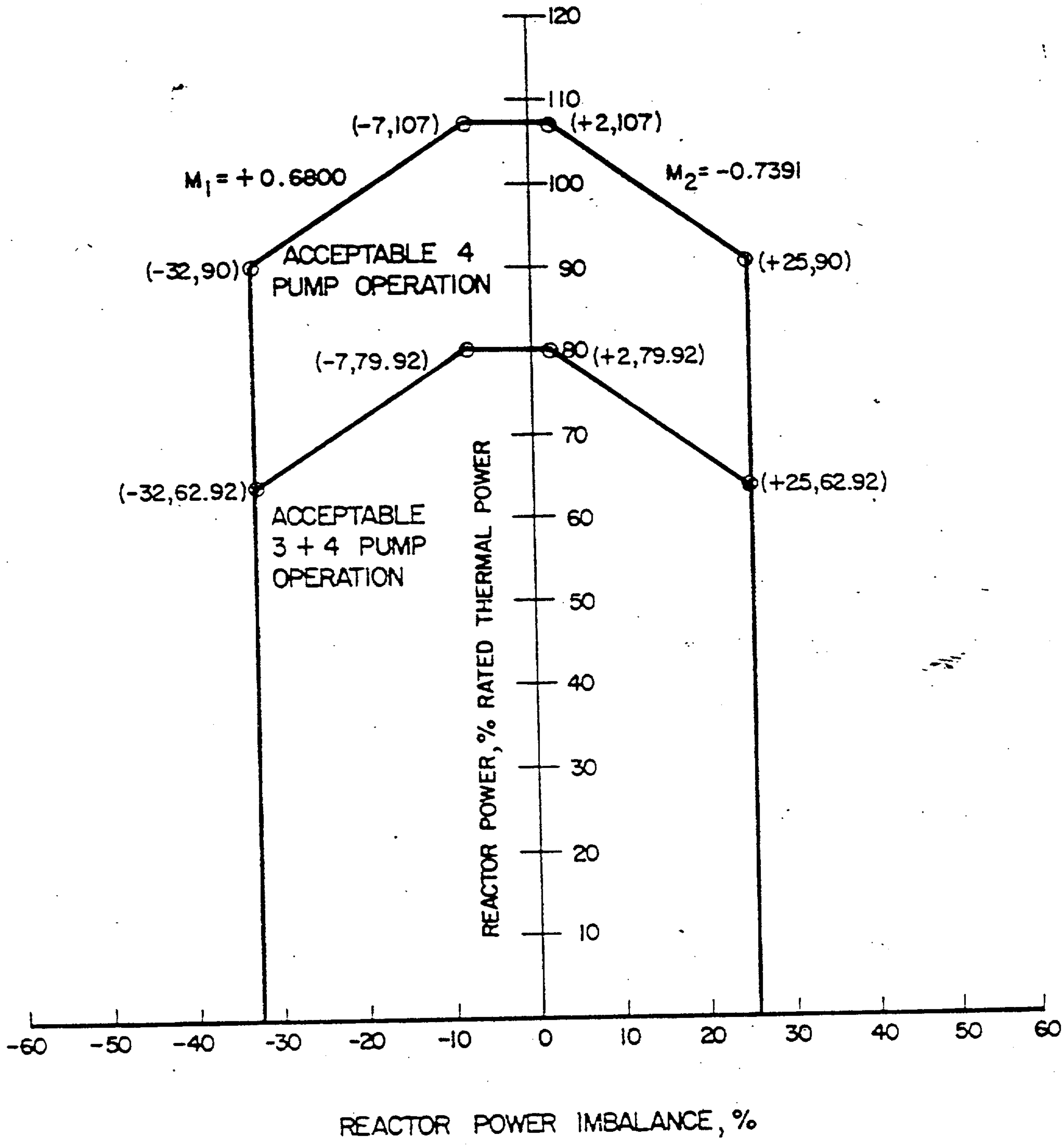

FIGURE 2.2-1

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

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip ≤ 618°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

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

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

1. Trip would occur when four reactor coolant pumps are operating if power is ≥ 107% and reactor flow rate is 100%, or flow rate is ≤ 93.45% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is > 79.92% and reactor flow rate is 74.7%, or flow rate is ≤ 70.09% and power is 75%.

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

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.07% for a 1% flow reduction.

RCS Pressure - Low, High, and Variable Low

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

During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2300 psig, has been established to maintain the system pressure below the safety limit, 2750 psig, for any design transient. The RCS Pressure-High trip is backed up by the pressurizer code safety valves for RCS over pressure protection and is, therefore, set lower than the set pressure for these valves, 2500 psig. The RCS Pressure-High trip also backs up the Nuclear Overpower trip.

The RCS Pressure-Low, 1800 psig, and RCS Pressure-Variable Low, (11.59 T_{out} °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 DNB correlation limits, protecting against DNB.

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

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.1.1.2 The SHUTDOWN MARGIN shall be $\geq$ 3.5% $\Delta k/k$.

APPLICABILITY: MODES 4 and 5.

ACTION:

With the SHUTDOWN MARGIN < 3.5% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 10 gpm of 11,600 ppm boric acid solution or its equivalent, until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.2.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.5% $\Delta k/k$:

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

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

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

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

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

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

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

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

ACTION:

MODES 1, 2, and 3:

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

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

MODE 4:

a. With the flow path from the concentrated boric acid storage system inoperable, restore the inoperable flow path to OPERABLE status within 72 hours or be borated to a SHUTDOWN MARGIN equivalent to 3.5% Δk/k at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

ACTION: (Continued)

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

SURVEILLANCE REQUIREMENTS

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

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

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

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4.2 At least one makeup pump shall be OPERABLE.

APPLICABILITY: MODE 4*

ACTION:

With no makeup pump OPERABLE, restore at least one makeup pump to OPERABLE status within one hour or be borated to a SHUTDOWN MARGIN equivalent to 3.5% $\Delta k/k$ at $200^{\circ}F$ and be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

*With RCS pressure $\geq$ 150 psig.

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:

MODES 1, 2, and 3:

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

MODE 4:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within 72 hours or be borated to a SHUTDOWN MARGIN equivalent to 3.5% $\Delta k/k$ at 200°F within the next 6 hours; restore at least one boric acid pump to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

BORATED WATER SOURCES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

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

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

1. A minimum contained borated water volume of 6,730 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. The concentrated boric acid storage system and associated heat tracing with:

1. A minimum contained borated water volume of 6,730 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 contained borated water volume of between 415,200 and 449,000 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:

MODES 1, 2, and 3:

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

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

MODE 4:

a. With the concentrated boric acid storage system inoperable, restore the storage system to OPERABLE status within 72 hours or be

ACTION: (Continued)

borated to a SHUTDOWN MARGIN equivalent to 3.5% $\Delta k/k$ at 200°F within the next 6 hours; restore the concentrated boric acid storage system to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

b. With borated water storage tank inoperable, restore the tank to OPERABLE status within one hour or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.9 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1. Verifying the boron concentration in each water source,

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

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

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

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT - SAFETY AND REGULATING ROD GROUPS

LIMITING CONDITION FOR OPERATIONS

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

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

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

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

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

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

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

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

*See Special Test Exceptions 3.10.1 and 3.10.2.

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 $K_{eff} \geq 1.0$.

REGULATING ROD INSERTION LIMITS

SURVEILLANCE REQUIREMENTS

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

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

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

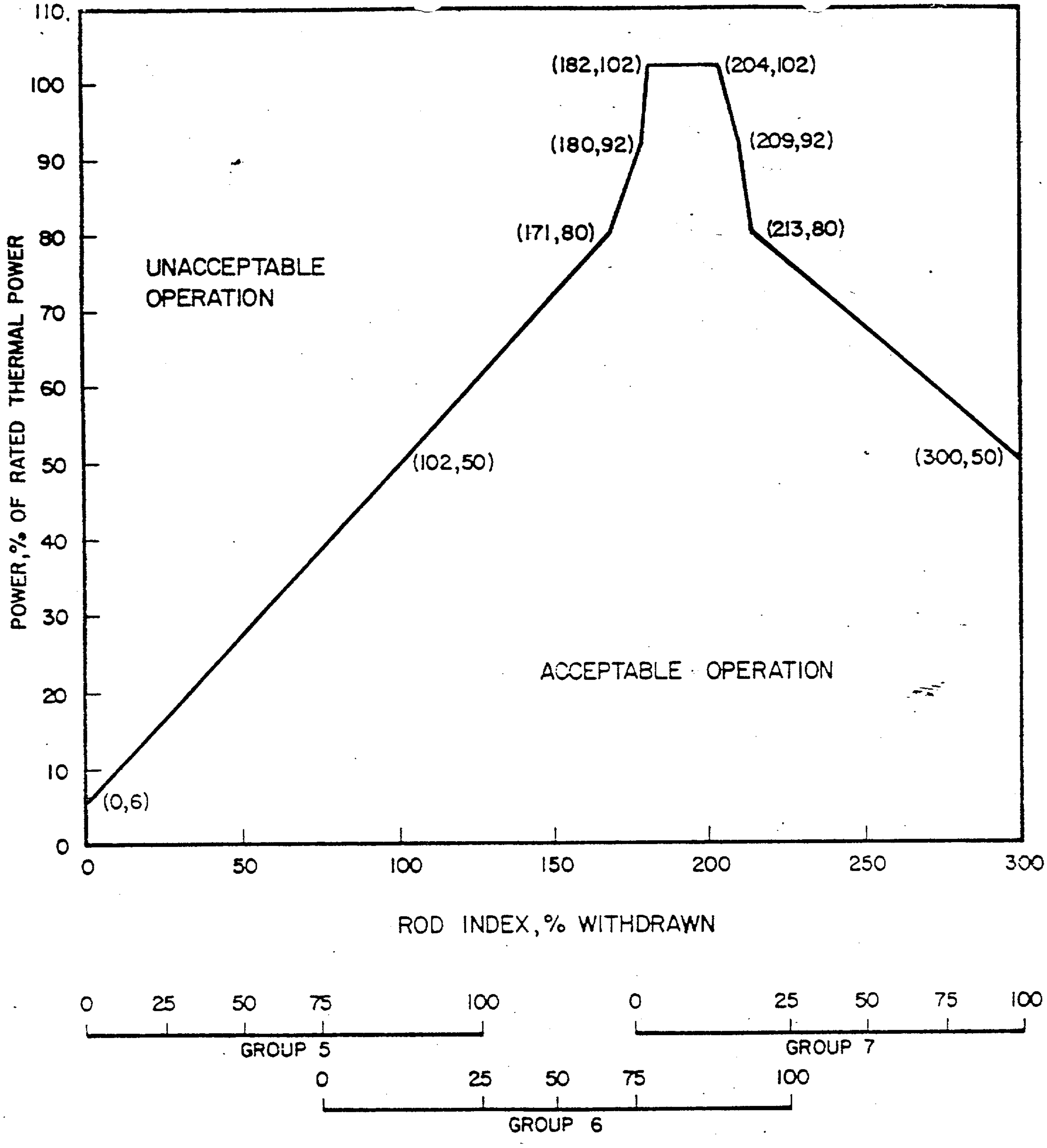

FIGURE 3.1-1

REGULATING ROD GROUP INSERTION LIMITS FOR 4 PUMP OPERATION FROM 0 EFPD TO 50 (+10/−0) EFPD

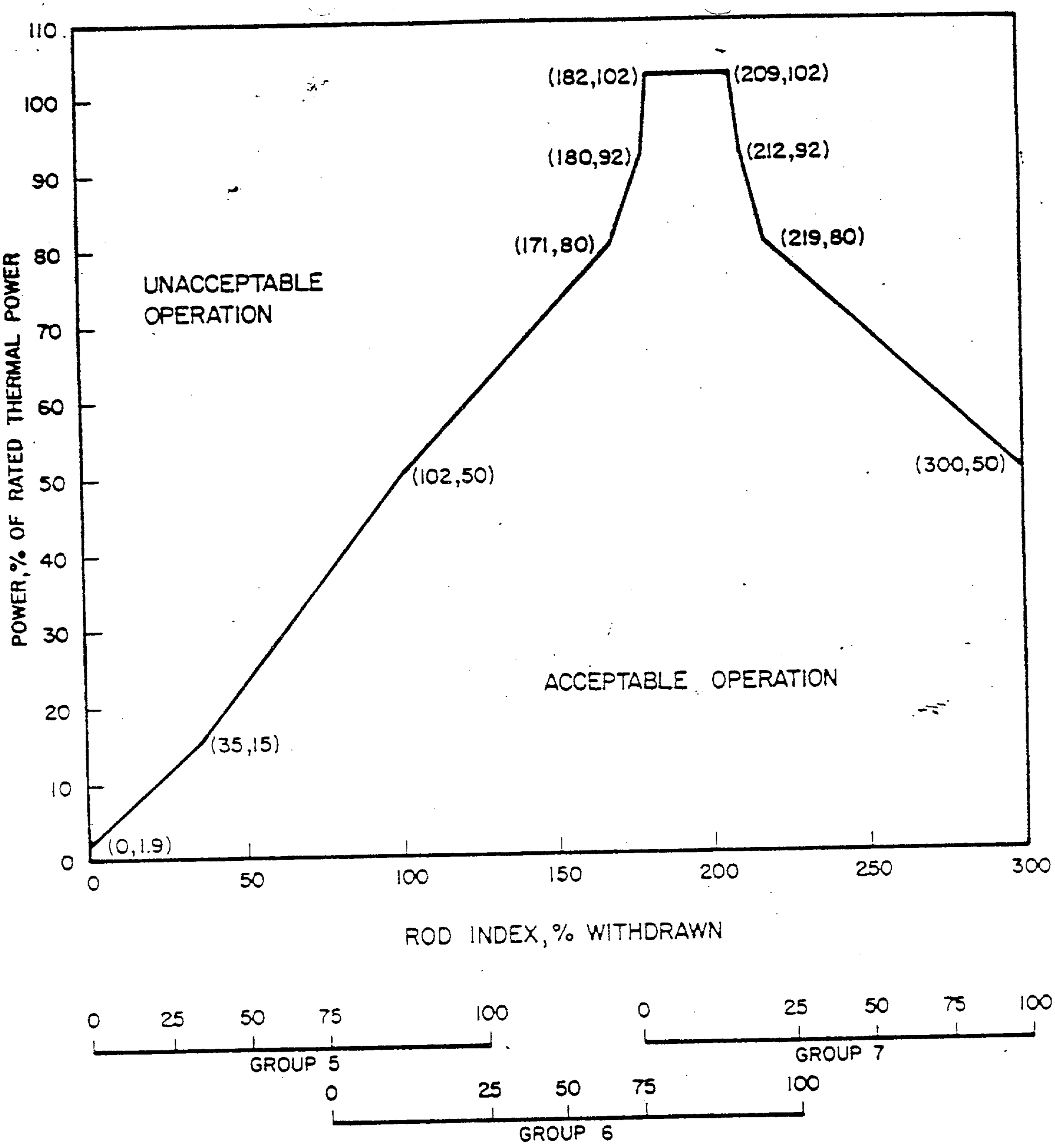

FIGURE 3.1-1a

REGULATING ROD GROUP INSERTION LIMITS FOR 4 PUMP OPERATION FROM 50 (+10/-0) TO 270 ± 10 EFPD

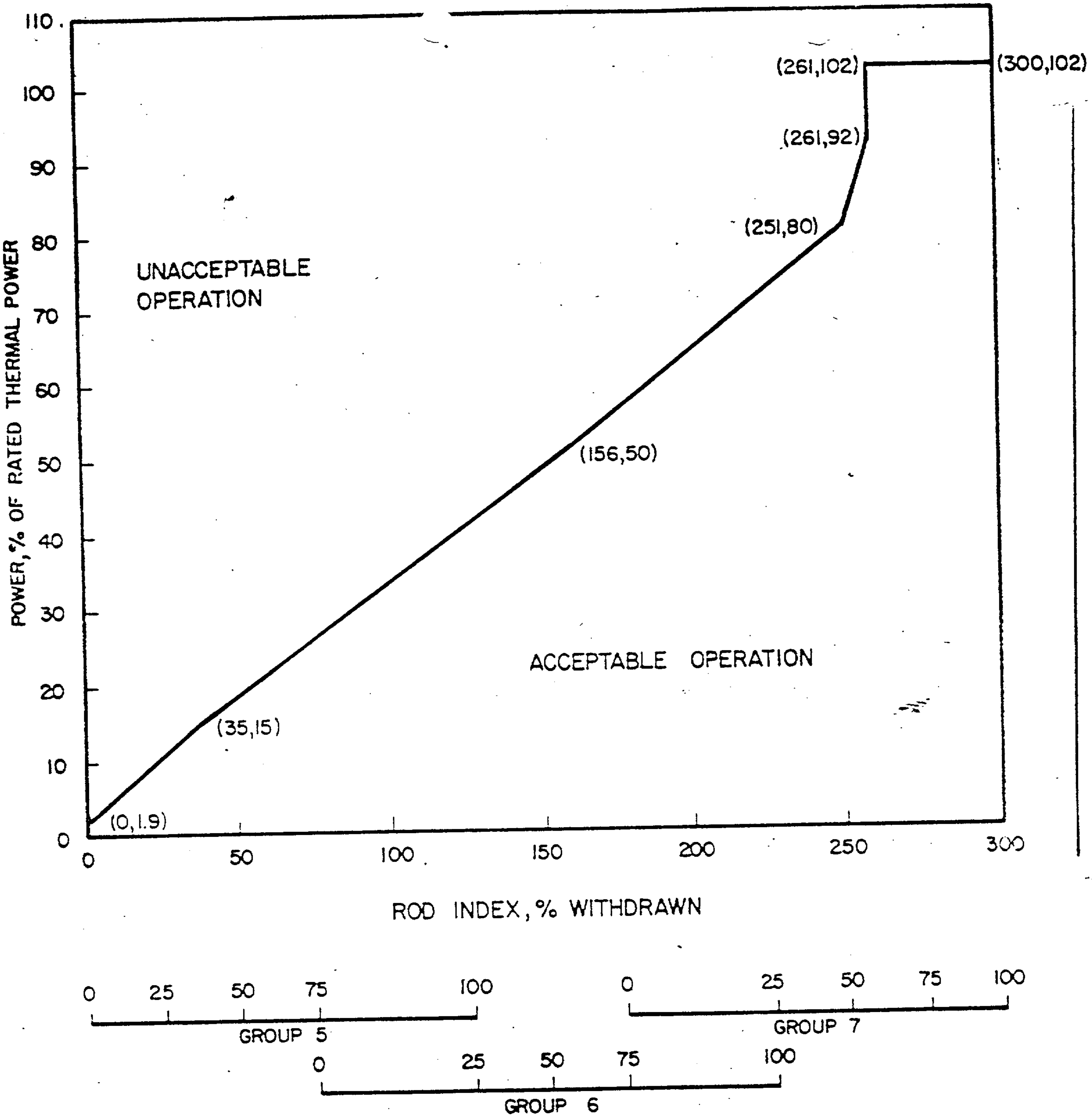

FIGURE 3.1-2

REGULATING ROD GROUP INSERTION LIMITS FOR 4 PUMP OPERATION AFTER 270 ± 10 EFPD

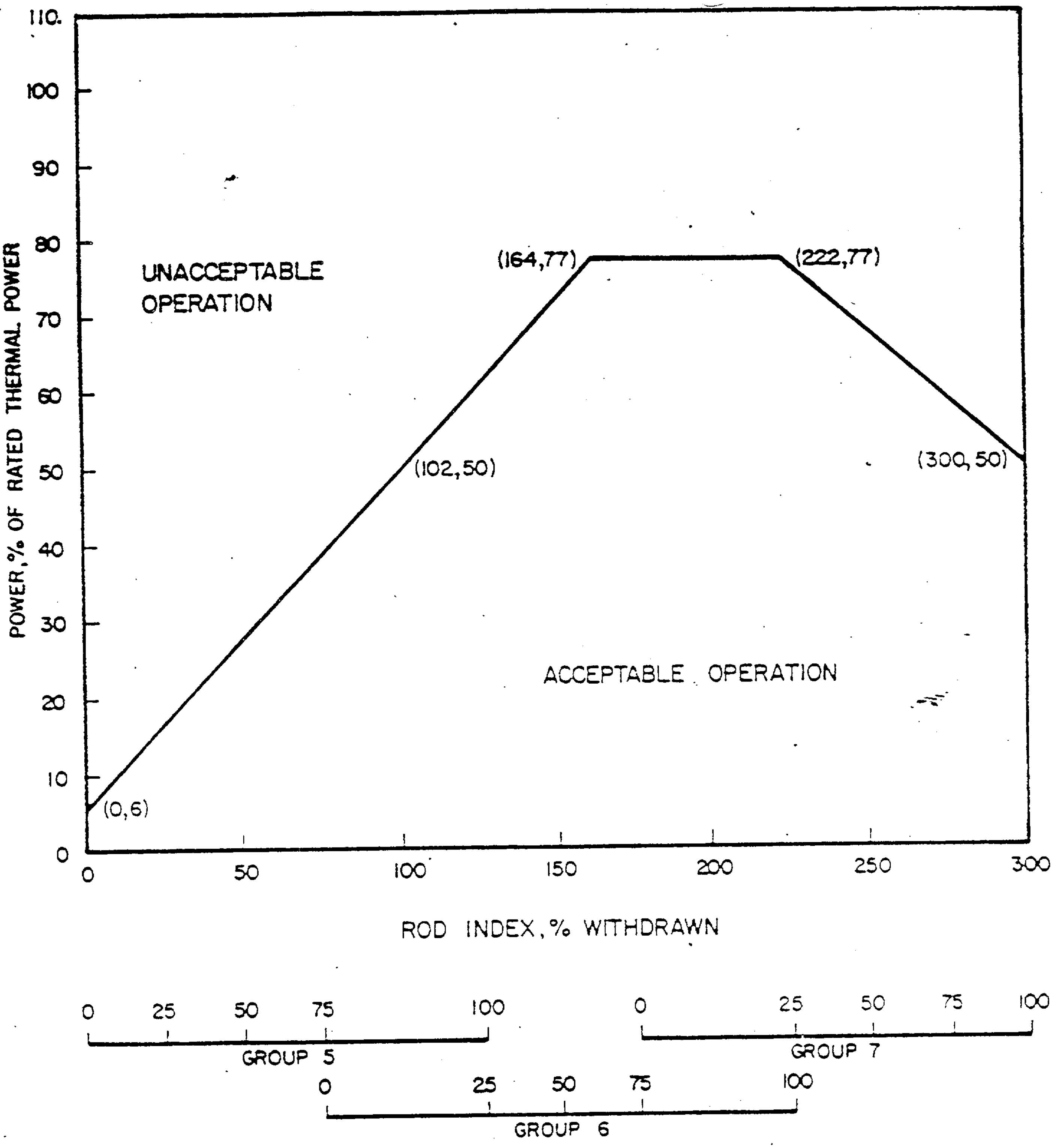

FIGURE 3.1-3

REGULATING ROD GROUP INSERTION LIMITS FOR 3 PUMP OPERATION FROM 0 EFPD TO 50 (+10/-0) EFPD

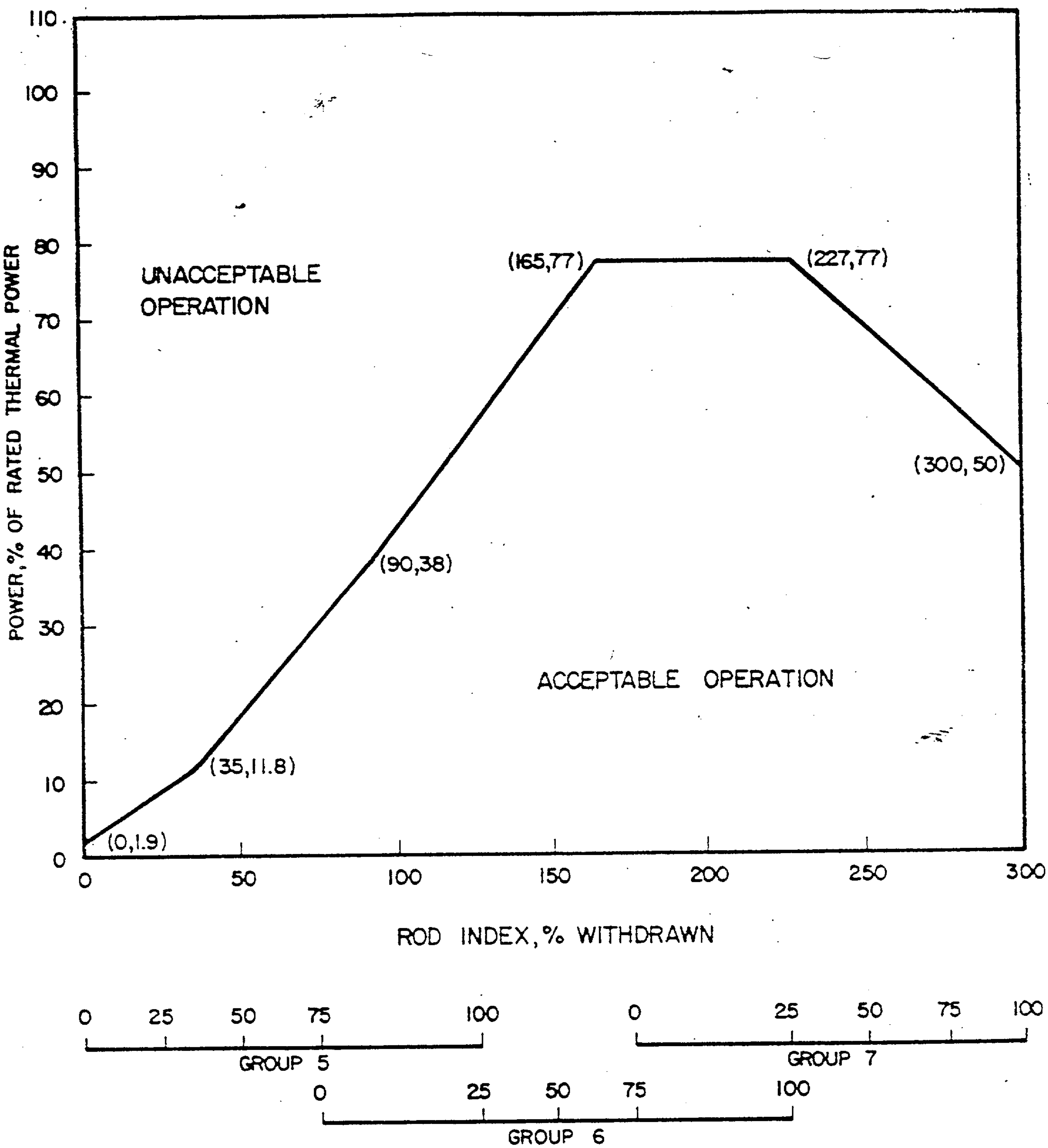

FIGURE 3.1-3a

REGULATING ROD GROUP INSERTION LIMITS FOR 3 PUMP OPERATION FROM 50 (+10/−0) TO 270 ± 10 EFPD

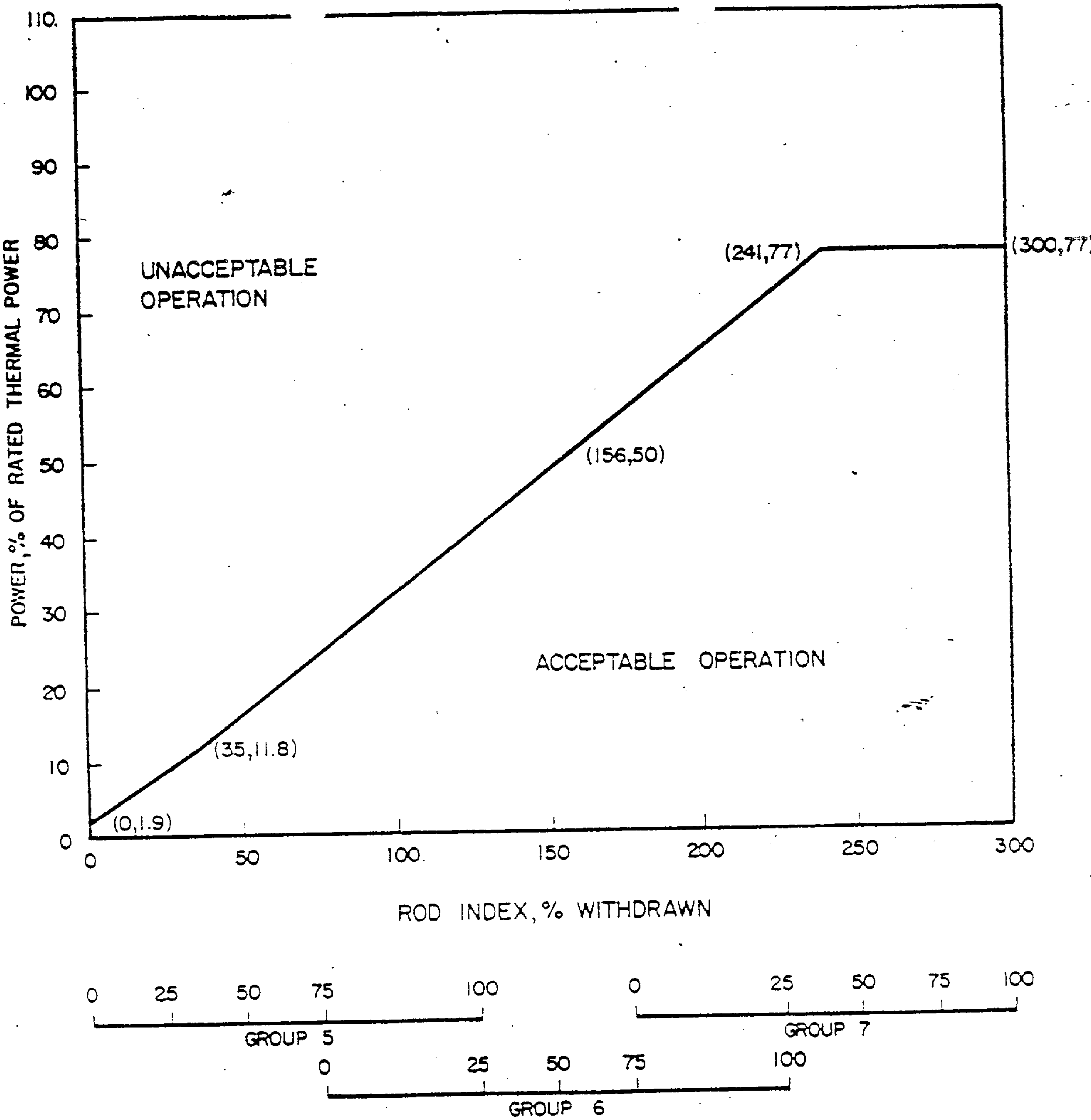

FIGURE 3.1-4

REGULATING ROD GROUP INSERTION LIMITS FOR 3 PUMP OPERATION AFTER 270 ± 10 EFPD

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2*.

ACTION:

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

a. Restore the axial power shaping rod group to within the limits within 2 hours, or

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

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

SURVEILLANCE REQUIREMENTS

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

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

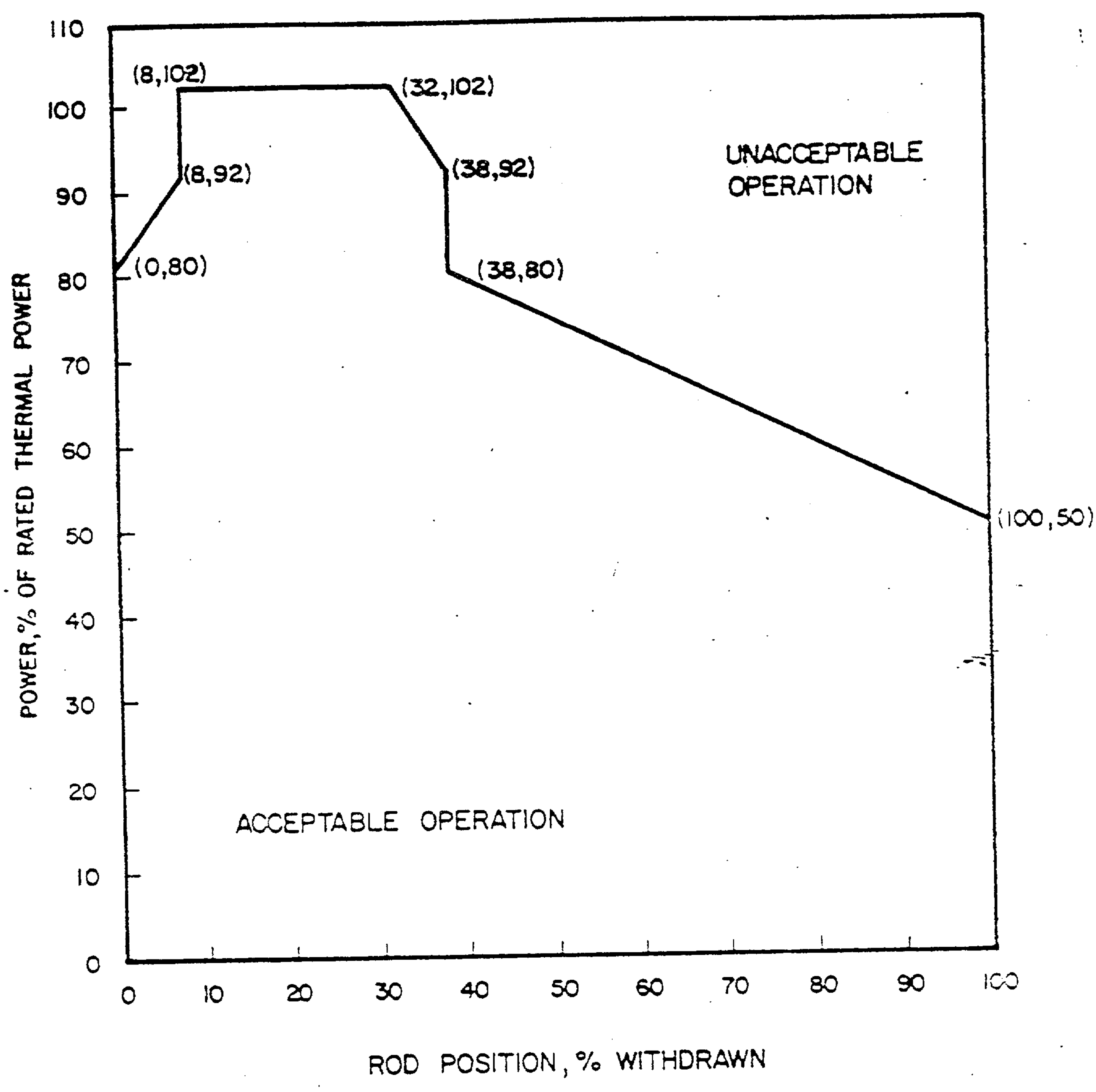

FIGURE 3.1-9

AXIAL POWER SHAPING ROD GROUP INSERTION LIMITS
FROM 0 EFPD TO 50 (+10/-0) EFPD

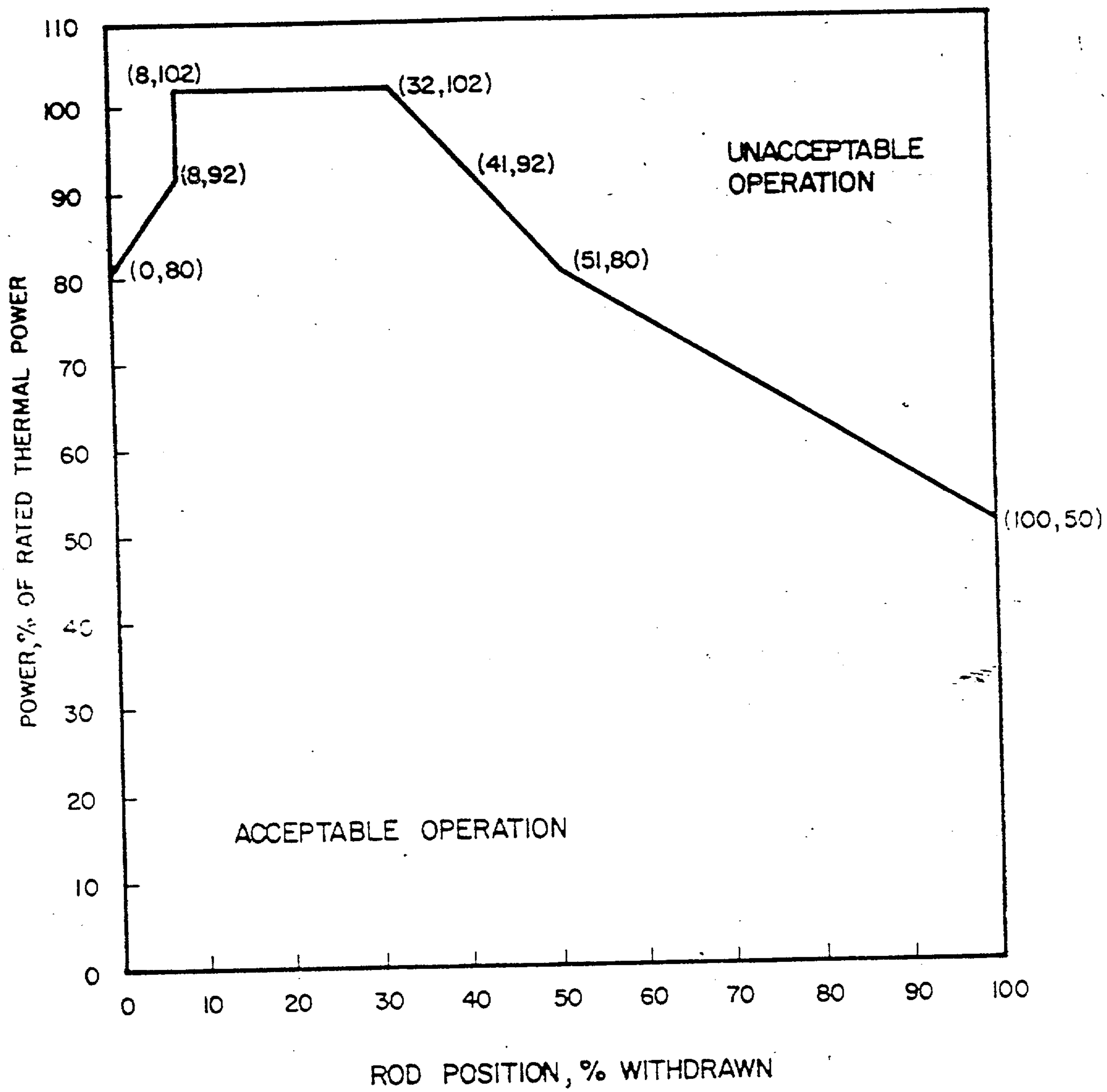

FIGURE 3.1-9a

AXIAL POWER SHAPING ROD GROUP INSERTION LIMITS
FROM 50 (+10/-0) TO 270 ± 10 EFPD

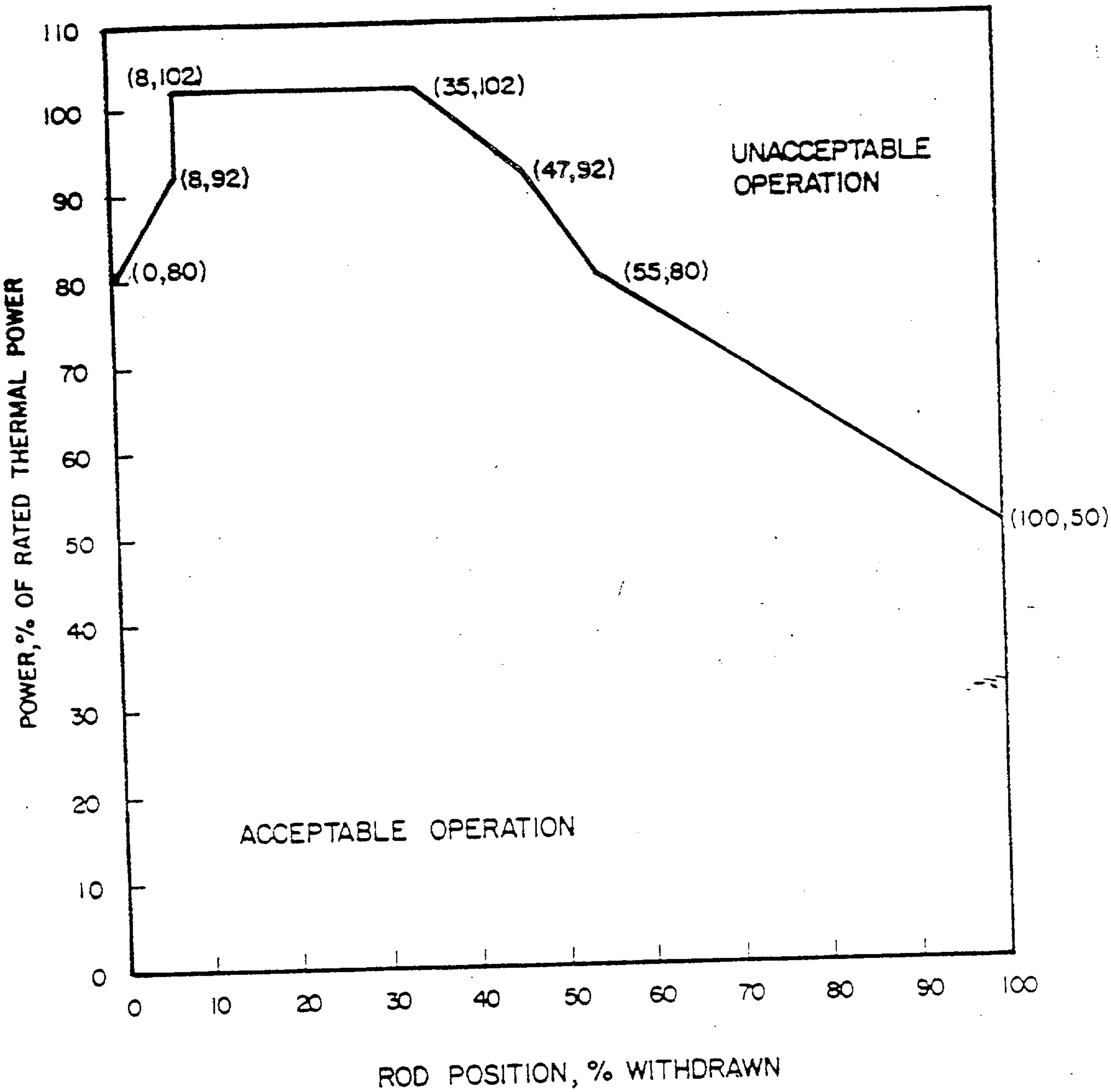

FIGURE 3.1-10

AXIAL POWER SHAPING ROD GROUP
INSERTION LIMITS AFTER 270 ± 10 EFPD

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

---

*See Special Test Exception 3.10.1.

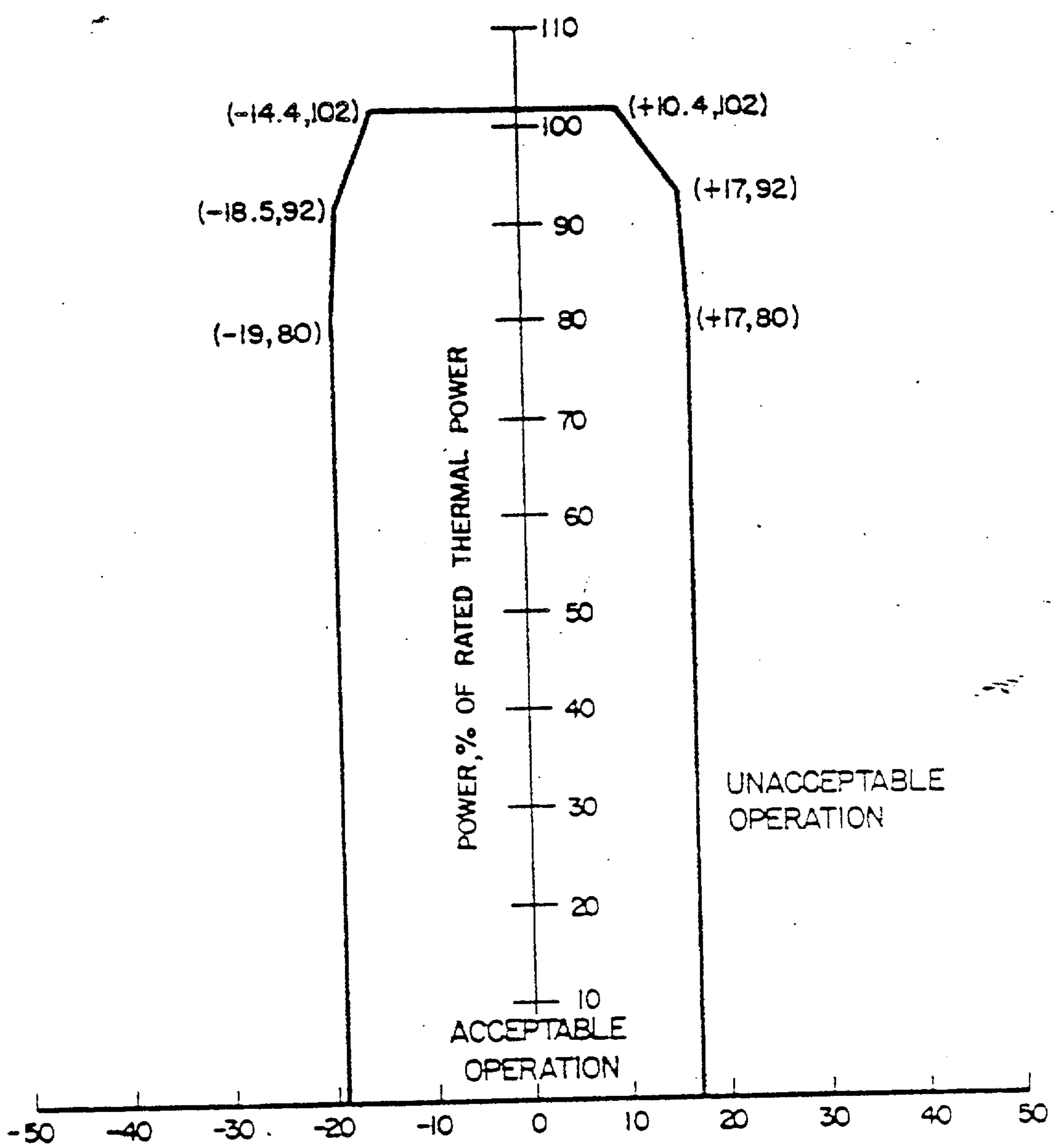

FIGURE 3.2-1

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 EFPD TO 50(+10/-0) EFPD

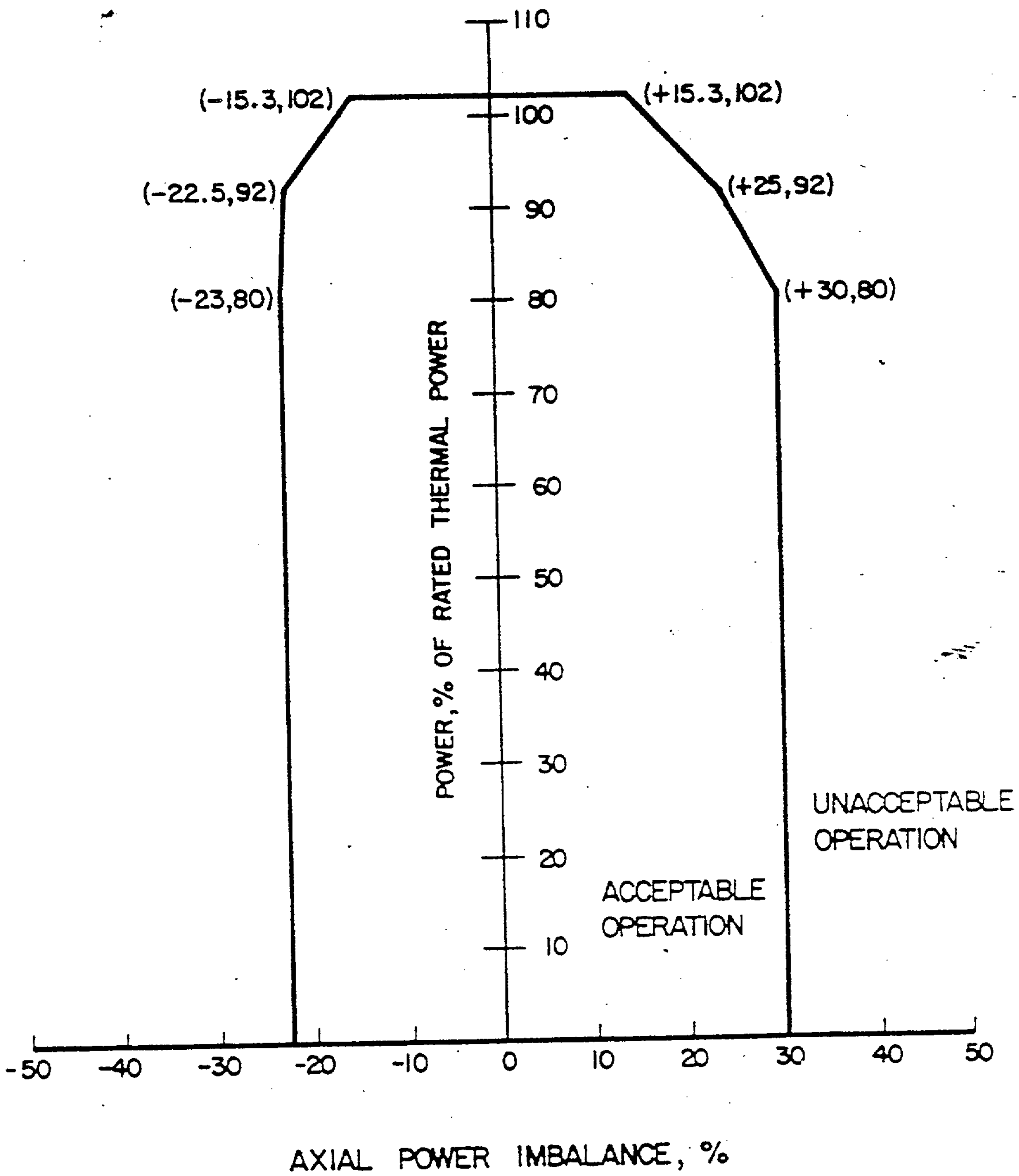

FIGURE 3.2-1a

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 50 (+10/-0) TO 270±10 EFPD

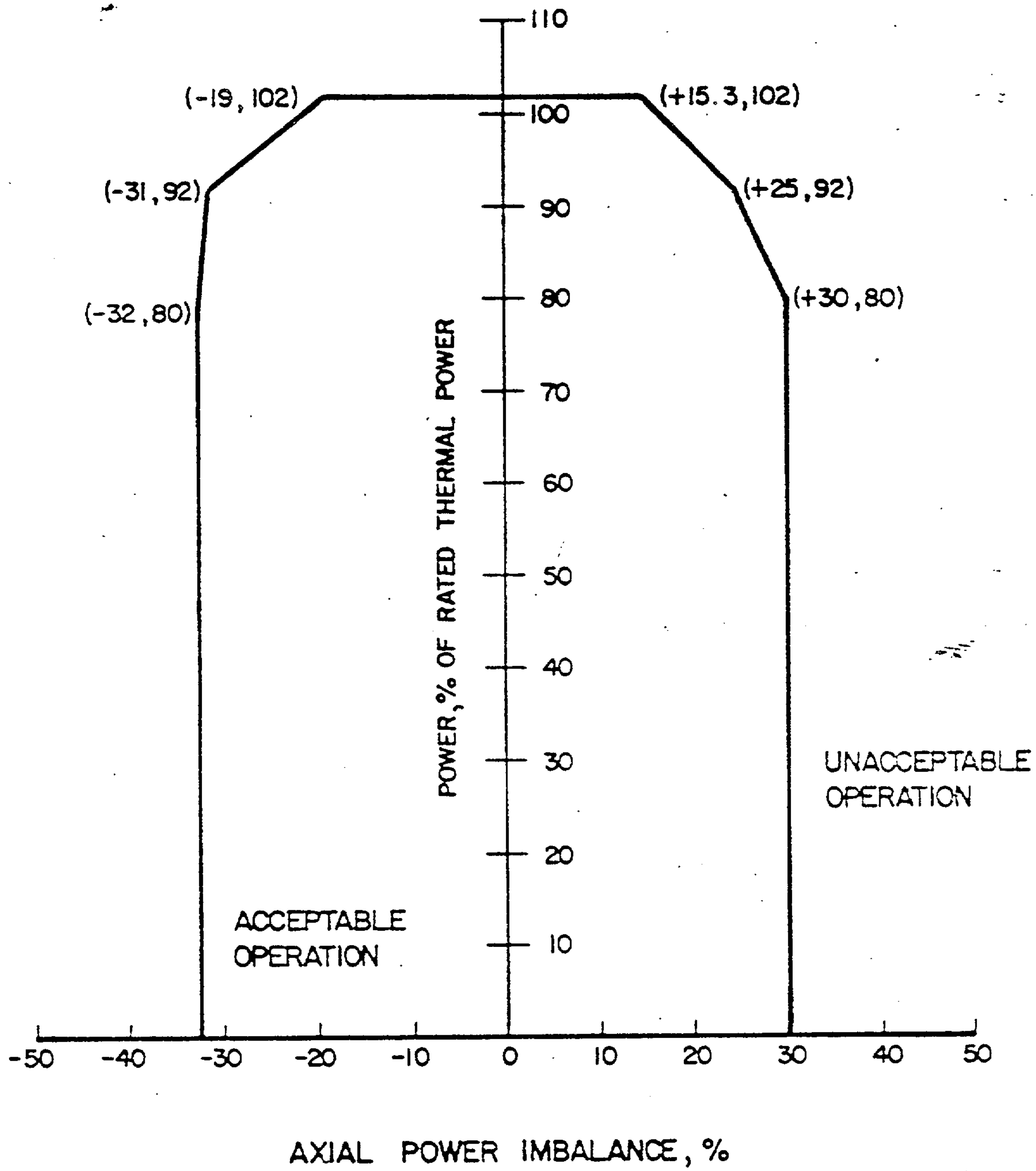

FIGURE 3.2-2

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION AFTER 270 ± 10 EFPD

POWER DISTRIBUTION LIMITS

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

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

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

APPLICABILITY: MODE 1.

ACTION:

With F_Q exceeding its limit:

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

b. Demonstrate through in-core mapping that F_Q is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 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:

ACTION STATEMENTS (Continued)

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

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

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

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

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

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

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

a. Within 1 hour:

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

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

b. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for 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.

TABLE 3.3-2

REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

Functional Unit	Response Times
1. Manual Reactor Trip	Not Applicable
2. Nuclear Overpower*	$\leq$ 0.326 seconds
3. RCS Outlet Temperature - High	Not Applicable
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE*	$\leq$ 1.79 seconds
5. RCS Pressure - Low	$\leq$ 0.5 seconds
6. RCS Pressure - High	$\leq$ 0.5 seconds
7. Variable Low RCS Pressure	Not Applicable
8. Nuclear Overpower Based on RCPPMs*	$\leq$ 0.47 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.

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

MODES 1 and 2:

a. With one reactor coolant pump not in operation, STARTUP and POWER OPERATION may be initiated and may proceed provided THERMAL POWER is restricted to less than 79.92% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

1. Nuclear Overpower

MODES 3, 4, and 5:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

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* See Special Test Exception 3.10.3.

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SURVEILLANCE REQUIREMENTS (Continued)

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

CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

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

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

1. Starting (unless already operating) each unit from the control room,
2. Verifying that each unit operates for at least 15 minutes, and
3. Verifying a cooling water flow rate of $\geq$ 500 gpm to each unit cooler.

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

3/4.1.1 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.

The most restrictive condition for MODES 4 and 5 occurs at BOL, and is associated with deboration due to inadvertent injection of sodium hydroxide. The higher requirement for these modes insures the accident will not result in criticality.

3/4.1.1.2 BORON DILUTION

A minimum flow rate of at least 2700 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual through the Reactor Coolant System in the core during boron concentration reductions in the Reactor Coolant System. A flow rate of at least 2700 GPM will circulate an equivalent Reactor Coolant System volume of 12,000 cubic feet in approximately 30 minutes. The reactivity change rate associated with boron concentration reduction will be within the capability for operator recognition and control.

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance requirement for measurement of the MTC each fuel cycle are adequate to confirm the MTC value since this coefficient changes slowly due principally to the reduction in RCS boron concentration associated with fuel burnup. The confirmation that the measured MTC value is within its limit provides assurance that the coefficient will be maintained within acceptable values throughout each fuel cycle.

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure 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 to 3.5% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs from full power equilibrium xenon conditions and requires either 6,730 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 47,698 gallons of 2,270 ppm borated water from the borated water storage tank.

The requirements for a minimum contained volume of 415,200 gallons of borated water in the borated water storage tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 3.5% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 300 gallons of 11,600 ppm boron from the boric acid storage system or 1,608 gallons of 2,270 ppm boron from the borated water storage tank. To envelop future cycle BWST contained borated water volume requirements, a minimum volume of 13,500 gallons is specified.

The contained water volume limits include allowance for water not available because of discharge line location and other physical characteristics. The limits on contained water volume and boron concentration ensure a pH value of between 7.2 and 11.0 of the solution sprayed within the containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chlorides and caustic stress corrosion cracking on mechanical systems and components.

The OPERABILITY of one boron injection system during REFUELING ensures that this system is available for reactivity control while in MODE 6.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section (1) ensure that acceptable power distribution limits are maintained, (2) ensure that the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effect of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. For example, misalignment of a safety or regulating rod requires a restriction in THERMAL POWER. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the safety analysis.

The position of a rod declared inoperable due to misalignment should not be included in computing the average group position for determining the OPERABILITY of rods with lesser misalignments.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analyses. Measurement with T_{avg} $\geq$ 525°F and with reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The limitation on Axial Power Shaping Rod insertion is necessary to ensure that power peaking limits are not exceeded.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 46 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated November 16, 1981, Florida Power Corporation (FPC or the licensee) requested a license amendment to the Appendix A Technical Specifications (TSs) of Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3) (Ref. 1). The proposed changes to the TSs address CR-3 Cycle 4 operation. The changes are supported in BAW-1684, "Crystal River Unit 3 - Cycle 4 Reload Report" (Ref. 2). This report provides the required analyses to support the operation of CR-3 Cycle 4 at the licensed rated core power level of 2544 MWT. These analyses employ analytical techniques and design bases established in NRC-approved reports.

2.0 THERMAL HYDRAULIC DESIGN EVALUATION

The CR-3 Cycle 4 core contains 177 fuel assemblies (FAs), each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one in-core instrument guide tube. Retainer assemblies will be on 64 assemblies which contain Burnable Poison Rod Assemblies (BPRAs) and two assemblies which contain regenerative neutron sources. The maximum fuel assembly burnup at end of Cycle 4 (EOC-4) is predicted to be 33,458 MWd/MtU.

We reviewed the thermal hydraulic design of the reload core to determine that it had been accomplished using acceptable methods and would provide an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients. The Cycle 4 core has been designed with an increased cycle lifetime of 350 effective full power days (EFPDs) and incorporates BPRAs to aid in reactivity control.

The thermal hydrualic evaluation of the reload Cycle 4 was performed utilizing TEMP (Ref. 26) and the B&W-2 critical heat flux (CHF) correlation (Ref. 27) as was done in Cycle 3 reload analysis. The licensed power level for CR-3 Cycle 4 operation is 2544 MWt. This power level was approved during Cycle 3 operations

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by Amendment No. 42 dated July 21, 1981. The thermal hydraulic design calculations in support of Cycle 4 operation assumed a power level of 2568 MWt (same as for Cycles 2 and 3) for consistency with other Babcock and Wilcox (B&W) plants. The main differences between Cycle 4 and the Reference Cycle 3 are discussed below. The Cycle 3 and 4 reference design conditions are summarized in Table 1 of this Safety Evaluation.

2.1 CORE BYPASS FLOW

The maximum core bypass flow in Cycle 3 was 10.4% (Ref. 28). For Cycle 4 operation, 64 BPRAs will be inserted, leaving 44 open assemblies, resulting in a decrease in calculated maximum core bypass flow to 8.1% (i.e., net increase in core flow).

2.2 BPRA RETAINERS

The retainers added to provide positive hold-down of BPRAs introduce a small departure from nucleate boiling ratio (DNBR) penalty discussed in Reference 29. However, the increase in core flow due to the BPRA insertion (Section 2.1 above) more than compensates for the decrease in DNBR due to the BPRA retainers.

2.3 ROD BOW DNBR PENALTY

The rod bow DNBR penalty applicable to Cycle 4, according to the licensee, was calculated using approved methods (Ref. 30). The burnup used to calculate the penalty was the highest batch 4C assembly burnup, 33,458 MWd/MtU. The resulting net rod bow penalty after inclusion of the 1% departure from nucleate boiling (DNB) credit for the flow area reduction hot channel factor is 3.5%. However, according to the licensee this rod bow penalty of 3.5% is offset by the 10.2% DNBR margin included in trip setpoints and operating limits (Ref. 28). The margin was incorporated to provide the flexibility for future cycle designs to avoid the potential need for revising setpoints on a cycle-by-cycle basis. Thus no power penalty for rod bowing is required for Cycle 4. This is acceptable to the NRC staff.

TABLE 1

THERMAL-HYDRAULIC DESIGN CONDITIONS

	Cycle 3	Cycle 4
Licensee power level, MWt	2544	2544
Design power level, MWt(a)	2568	2568
System pressure, psia	2200	2200
Reactor coolant design flow, GPM	352000	352000
Reactor coolant flow, % design	106.5	106.5
Core bypass flow % design	10.4	8.1
Actual core flow, GPM	315392	323488
Reference design radial x local power peaking factor, $F_{\Delta H}$	1.71	1.71
Reference design axial flow shape	1.5 cosine	1.5 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Densified active length, in.(a)	140.2	140.2
Average heat flux at 100% power, Btu/h-ft^2	176×10^3	176×10^3
Maximum heat flux at 100% power, Btu/h-ft^2	452×10^3	452×10^3
CHF correlation	BAW-2	BAW-2
Minimum DNBR, % power	1.98(112)	2.05(112)
Limiting Transient* (one pump coast down) MDNBR	1.75	1.69
Peak allowable linear heat rate (kw/ft)	14.91	14.91

(a) Used in analysis.

* This difference in MDNBR value for Cycles 3 and 4 exists because of the revised setpoints to compensate the increased RPS instrument errors.

2.4 LOSS-OF-COOLANT FLOW TRANSIENTS

The limiting transient (one pump coastdown) minimum DNBR value was 1.75 for Cycle 3 compared to a minimum DNBR value of 1.69 for Cycle 4 (Ref. 28). This difference in minimum DNBR value for Cycles 3 and 4 exists because of the revised setpoints to compensate the increased Reactor Protection System (RPS) instrument errors. The Cycle 4 minimum DNBR value of 1.69 represents a 30% DNBR margin to the correlation limit of 1.30, and is therefore acceptable.

2.5 CONCLUSION

We conclude from the examination of Cycle 4 core thermal and thermal hydraulic properties, with respect to approved previous cycle values and with respect to the Final Safety Analysis Report (FSAR) values, that the analysis supports the operation of CR-3 during Cycle 4.

3.0 EVALUATION OF TRANSIENT AND ACCIDENT ANALYSES

The licensee examined all accidents and transients analyzed in the FSAR and concluded that they are bounded by the FSAR and/or the fuel densification report, "BAW-1397".

The key parameters that have the greatest effect on determining the outcome of a transient can typically be classified in three major areas: core thermal parameters, thermal-hydraulic parameters, and kinetics parameters.

The core thermal properties used in the FSAR analysis were design operating values based on calculational values plus uncertainties. The Cycle 4 thermal hydraulic maximum design conditions are compared to the previous Cycle 3 values in Table 1. These parameters are common to all the accidents considered and as shown in Table 1, the only change is in the minimum DNBR value. The change is in the conservative direction. Also, a comparison of the key kinetics parameters of the FSAR , Cycles 1, 3 and Cycle 4 is shown in Table 2. All changes are on the conservative side.

Based on the above evaluation we have concluded that the Cycle 4 reload for CR-3 is acceptable with respect to transient and accident analysis.

4.0 EVALUATION OF FUEL SYSTEM DESIGN

4.1 FUEL ASSEMBLY MECHANICAL DESIGN

The sixty-eight Babcock and Wilcox (B&W) Mark-B4 15x15 fuel assemblies loaded as Batch 6 at the end of Cycle 3 (EOC 3) are mechanically interchangeable with Batches 4A, 4C and 5 fuel assemblies previously loaded at CR-3. The Mark-B4 fuel assembly has been previously approved (Ref. 3) by the NRC staff and is utilized in other B&W nuclear steam supply systems. Two assemblies will contain regenerative neutron sources, and retainers will be used to contain these sources as well as a number of BPRAs. Justification for the design and use of the neutron source retainer is described in the "Burnable Poison Rod Assembly Retainer Design Report" (Ref. 4). A discussion of the burnable poison rods themselves is presented in Section 4.1.1 of this evaluation.

4.1.1 REACTIVITY CONTROL SYSTEM

In addition to the permanent reactivity control system (soluble boron, control rods, and axial shaping rods), sixty-four BPRAs are being added to control reactivity changes due to fuel burnup and fission product buildup. The BPRAs are normally removed from the reactor at the end of first cycle and reinserted only for extended cycle operation, such as that proposed for Cycle 4. In April 1978, two BPRAs were accidentally ejected from the core of CR-3 (Ref. 5). The ejected BPRAs were carried out of the reactor vessel by the coolant flow to the steam generator, where significant damage to the steam generator tube ends resulted. B&W determined

Table 2. Comparison of Key Parameters for Accident Analysis

Parameter	FSAR, densif'n value	Cycle 1	Predicted Cycle 4 Value	Cycle 3
BOL Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.17	-1.47 (268 EFPD)	-1.55	-1.52
EOL Doppler coeff, 10^{-5} $\Delta k/k/°F$	-1.30	-1.66 (510 EFPD)	-1.71	-1.6
BOL moderator coeff, 10^{-4} $\Delta k/k/°F$	0(a)	-0.75 (268 EFPD)	-0.52	-0.3
EOL moderator coeff, 10^{-4} $\Delta k/k/°F$	-4.0	-2.42 (510 EFPD)	-2.89	-2.63
All-rod bank worth at BOL, HZP, % $\Delta k/k$	12.9	9.12 (268 EFPD)	9.583	<9.37
Boron reactivity worth (HFP), ppm/1% $\Delta k/k$	100	101	111	108
Max. ejected rod woth (HFP), % $\Delta k/k$	0.65	0.55	0.564	.49
Dropped rod worth (HFP), % $\Delta k/k$	0.40	0.20	0.20	.2
Initial boron conc'n (HFP), ppm	1150	795	1090	1185

(a) $+0.50 \times 10^{-4}$ $\Delta k/k/°F$ was used for the moderator dilution accident.

(b) -3.0×10^{-4} $\Delta k/k/°F$ was used for the steam line failure analysis and dropped rod accident analysis.

that the ejection of the BPRAs from the core resulted from fretting wear in the holddown latching mechanism. In order to avoid similar problems at other plants, B&W redesigned and replaced the BPRA holddown mechanism on all operating B&W cores. The NRC staff has generically approved (Ref. 6) the new design. We therefore conclude that changes to the core reactivity control system have been adequately considered for Cycle 4 operation.

4.2 FUEL ROD DESIGN

Although all batches in CR-3 Cycle 4 utilize the same Mark-B4 fuel, the Batch 6 assemblies incorporate a slightly higher average enrichment. Fixty-six assemblies contain 2.62 w/o U-235 and are designated Batch 6A. This enrichment is identical to that in the previous Batch 5 fuel. The remaining twelve assemblies contain 2.95 w/o U-235 and are designated Batch 6B. The undensified active fuel length has also been increased slightly in the Batch 6 fuel. We regard these changes as minor and therefore acceptable.

4.2.1 CLADDING COLLAPSE

The licensee has stated that the cladding collapse analysis in the Cycle 4 Reload Report is bounded by conditions previously analyzed with an approved generic creep collapse analysis. We conclude that cladding collapse has been appropriately considered for Cycle 4 operation.

4.2.2 CLADDING STRESS

The licensee has stated that the cladding stress analysis described in the Cycle 4 Reload Report is bounded by conditions previously analyzed for CR-3 or analyzed specifically for Cycle 4 conditions using methods and limits previously reviewed and approved by the NRC. We conclude that additional NRC staff review of the cladding stress analysis is unnecessary for Cycle 4 operation.

4.2.3 CLADDING STRAIN

The licensee has stated that the cladding strain analysis described in the Cycle 4 Reload Report is bounded by conditions previously analyzed for CR-3. We conclude that additional NRC staff review of the cladding strain analysis is also unnecessary for Cycle 4 operation.

4.2.4 ROD INTERNAL PRESSURE

Section 4.2 of the Standard Review Plan (Ref. 7) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those which may affect the operation of the fuel rod is the internal pressure limit. The acceptance criterion (SRP 4.2, Section II.A.1(f)) is that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified.

The licensee has stated that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 4. This analysis is based on the use of the B&W TAFY code (Ref. 8) rather than a newer B&W code called TACO-1 (Ref. 9). Although both of these codes have been approved for use in safety analysis, we believe (Ref. 10) that only the newer TACO code is capable of correctly calculating fission gas release (and therefore rod pressure) at very high burnups. Babcock and Wilcox has responded (Ref. 11) to this concern with an analytical comparison between both codes. In this response, they have stated that the internal fuel rod pressure predicted by TACO is lower than that predicted by TAFY for fuel rod exposures of up to 42,000 MWd/MtU. Although we have not examined the comparison, we note that the analyses exceed the expected exposure (33,458 MWd/MtU) in CR-3 at EOC-4 for all fuel assemblies. We conclude that the rod internal pressure limits have been adequately considered for Cycle 4 operation.

4.3 FUEL THERMAL DESIGN

The average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the Loss of Coolant Accident (LOCA) analysis (Section 7.2 of the Reload submittal) are also calculated with the TAFY code (Ref. 8). The licensee has stated that the fuel temperature and pin pressure data used in the generic LOCA analysis are conservative compared with those calculated for Cycle 4 at CR-3.

As previously mentioned in Section 4.2.4 of this evaluation, B&W currently has several fuel performance codes including TAFY (Ref. 8), TACO-1 (Ref. 9) and a recently submitted code called TACO-2 (Ref. 12). The first two are approved and could be used to calculate the LOCA initial conditions. The older TAFY code has been used for the Cycle 4 LOCA analysis. Recent information (Ref. 13) indicates that the TAFY code predictions do not produce higher peak cladding temperatures than TACO-1 for all Cycle 4 conditions as suggested in Ref. 11. The issue involves calculated fuel rod internal gas pressures that are too low at beginning of life. The rod internal pressures are used to determine swelling and rupture behavior during LOCA. B&W has proposed (Ref. 14) a method of resolving this issue which we accepted (Ref. 15). The method involves the use of reduced LOCA kW/ft limits at low core elevations during the first 50 EFPDs of operation. The licensee has incorporated these changes into the CR-3 TSs to support the operation during Cycle 4. We have reviewed these changes and have found them acceptable. We conclude that the initial thermal conditions for LOCA analysis have been appropriately considered for Cycle 4 operation.

4.4 MATERIAL COMPATIBILITY

The chemical and material compatibility of possible fuel, cladding, and coolant interactions is unchanged from the previous cycle of operation. The impact of this issue on the operational safety of CR-3 need not be reconsidered for Cycle 4 operation.

4.5 OPERATING EXPERIENCE

B&W has accumulated operating experience with the Mark B 15x15 fuel assembly at all of the eight operating B&W 177-fuel-assembly plants. A summary of this operating experience as of March 31, 1981 is given on page 4-2 of Ref. 2.

4.5.1 GUIDE TUBE WEAR

Significant wear of Zircaloy control rod guide tubes has been observed in facilities designed by Combustion Engineering. Similar wear has also been reported in facilities designed by Westinghouse. In a letter dated June 13, 1978, we requested information from B&W on the susceptibility of the facilities designed by B&W to guide tube wear. The information provided by B&W in a letter dated January 12, 1979, was insufficient for us to conclude that guide tube wear was not a significant problem in B&W plants. This insufficiency was documented in our letter to B&W dated August 22, 1979.

Because guide tube wear could result in loss of control rod scram capability and also fuel assembly structural integrity, we considered this wear phenomenon a potential safety concern. Therefore, we requested (Ref. 16) additional information from the licensee on control rod guide tube wear. The licensee has provided a generic response to this request, the B&W Control Rod Guide Tube Wear Generic Report (Ref. 17), which is applicable to CR-3. We recently completed our review of this report and have found it acceptable (Ref. 18). The report provides information on post-irradiation examinations of guide tube wear in Oconee 1 and 3 and Rancho Seco spent fuel. The results of these measurements indicate that through-wall wear or excessive wall degradation will not likely occur during anticipated fuel residence time for rodded assemblies. On the basis of our generic evaluation, we conclude that guide tube wear has been adequately addressed for CR-3 during Cycle 4 and subsequent cycles meeting the conditions of Reference 18.

4.5.2 HOLDDOWN SPRING FAILURES

The upper end fitting of the B&W Mark-B4 fuel assembly contains a holddown spring to accommodate length changes due to thermal expansion and irradiation growth while providing a positive holddown force for the assembly. On May 14, 1980, a failed holddown spring was discovered by remote video inspection at Davis-Besse Unit 1 (Ref. 19). Further examination ultimately identified a total of 19 failed springs in the Davis-Besse Cycle 1 fuel assemblies. Subsequent examination of spent fuel assemblies at other B&W

reactors revealed a small number of similar failures at Oconee 1 (Ref. 20) as well as CR-3 (Ref. 21).

We have reviewed the B&W holddown spring failures as a generic issue (Ref. 22). The predominant mode of failure appears to have been fatigue-initiated cracking followed by stress corrosion crack propagation to failure of springs with an improper metallurgical condition (grain size). Based upon our review of information provided (a) in a meeting with Toledo Edison (Davis-Besse) in June and (b) responses to NRC staff questions issued to all B&W licensees in July, we believe that there is reasonable assurance that the holddown spring failures will not occur on as large a scale again, and that neither the potential for (a) loss of positive holddown force, (b) loose parts, nor (c) interference for normal control rod assembly movement constitute a significant safety hazard.

Nevertheless, because at least one failure (at CR-3) appears not to be related to material of improper metallurgical condition, and because some lateral and vertical motion of loose assemblies is theoretically possible under certain extreme conditions, we have concluded that further surveillance (e.g., video examination) of the assembly holddown springs should be carried out at the next refueling at each plant. The licensee committed to report the results of such an examination (Ref. 23) of all fuel assemblies at CR-3 for holddown spring damage during the previous outage, and no assemblies containing failed springs were reported.

On the basis of the fuel vendor's analysis of the consequences of operating with failed holddown springs, the completion of our generic evaluation of the problem, and the results of the licensee's examination of fuel assemblies at CR-3, we conclude that there is reasonable assurance that the holddown spring issue has been correctly analyzed and that this issue does not present a safety concern for future cycles of operation.

4.6 FUEL ROD BOWING

The licensee has stated that a fuel rod bowing penalty has been calculated with a method similar to that described in Reference 24. The rod bowing magnitude correlation used in that method is described in Reference 25, and we conclude that it adequately accounts for gap closure as a function of burnup in the Mark-B4 fuel design. The remaining input assumptions for the rod bowing analysis, and the manner in which the resultant rod bowing penalty is offset, are described in the Thermal Hydraulic Design section of this report.

4.6 CONCLUSIONS

We have reviewed the reload report for CR-3 Cycle 4 dealing with the fuel system design and find the analysis adequately supports operation for Cycle 4.

5.0 ENVIRONMENTAL CONSIDERATIONS

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

6.0 CONCLUSIONS

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) 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: December 4, 1981

References

1. Patsy Y. Baynard (FPC) letter to the Director, Office of Nuclear Reactor Regulation (NRC) dated November 16, 1981.

2. Crystal River Unit 3 Cycle 4 Reload Report, Babcock and Wilcox Company Report BAW-1684, June 1981.

3. R. W. Reid (NRC) letter to J. A. Hancock (FPC) on "Amendment 32" dated August 1, 1980.

4. BPRA Retainer Design Report, Babcock & Wilcox Company Report BAW-1496, May 1978.

5. W. P. Stewart (FPC) letter to C. Nelson (NRC) on "Crystal River Unit Three Status Report - May 1, 1978," dated May 4, 1978.

6. T. M. Novak (NRC) memorandum to E. L. Jordan (NRC) dated December 22, 1980.

7. Standard Review Plan, Section 4.2 (Rev. 1), "Fuel System Design," U.S. Nuclear Regulatory Commission Report NUREG-75/087.

8. C. D. Morgan and H. S. Kao, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis," Babcock and Wilcox Company Report BAW-10044, May 1972.

9. "TACO-Fuel Pin Performance Analysis," Babcock and Wilcox Company Report BAW-10087P-A, Rev. 2, August 1977.

10. D. F. Ross, Jr. (NRC) letter to J. H. Taylor (B&W) dated January 18, 1978.

11. J. H. Taylor (B&W) letter to P. S. Check (NRC) dated July 18, 1978.

12. Y. H. Hsii et al., "TACO-2: Fuel Pin Performance Analysis," Babcock and Wilcox Company Report BAW-10141P, January 1979.

13. R. O. Meyer (NRC) memorandum to L. S. Rubenstein (NRC) on "TAFY/TACO Fuel Performance Models in B&W Safety Analyses," dated June 10, 1980.

14. J. H. Taylor (B&W) letter to L. S. Rubenstein (NRC) dated September 5, 1980.

15. L. S. Rubenstein (NRC) letter to J. H. Taylor (NRC) dated October 28, 1980.

16. R. W. Reid (NRC) letter to W. P. Stewart (FPC) dated November 23, 1979.

17. J. H. Taylor (B&W) letter to D. G. Eisenhut (NRC) dated July 18, 1980 and transmitting Control Rod Guide Tube Wear Measurement Program (BAW-1623) dated June 1980.

18. L. S. Rubenstein (NRC) memorandum for R. L. Tedesco (NRC) on "Review of Topical Report BAW-1623," dated July 28, 1981.

19. T. D. Murray (Toledo Edison) letter to J. G. Keppler (NRC/Reg. III) dated May 23, 1980.

20. W. O. Parker, Jr. (Duke Power) letter to J. P. O'Reilly (NRC/Reg. II) dated June 6, 1980.

21. J. A. Hancock (FPC) letter to J. P. O'Reilly (NRC/Reg. II) dated May 29, 1980.

22. L. S. Rubenstein (NRC) memorandum for T. M. Novak (NRC) on "B&W Fuel Assembly Holddown Spring Failures," dated December 30, 1980.

23. P. Y. Baynard (FPC) letter to J. F. Stolz (NRC) dated March 30, 1981.

24. L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) on "Evaluation of Interim Procedure for Calculating DNBR Reduction Due to Rod Bow," dated October 18, 1979.

25. R. O. Meyer (NRC) memorandum to D. F. Ross (NRC) on "Revised Coefficients for Interim Rod Bowing Analysis," dated March 2, 1978.

26. TEMP-Thermal Enthalpy Mixing Program, BAW-10021, Babcock & Wilcox, Lynchburg, Virginia, April 1970.

27. Correlation of Critical Heat Flux in Bundle Cooled by Pressurized Water, BAW-10000A, Babcock and Wilcox, Lynchburg, Virginia, May 1976.

28. P. Y. Baynard (FPC) letter to H. R. Denton (NRC), "Response to NRC Questions", dated November 19, 1981.

29. BPRA Retainer Design Report, BAW-1496, Babcock and Wilcox, Lynchburg, Virginia, May 1978.

30. L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W), "Evaluation of Interim Procedure for Calculating DNBR Reductions Due to Rod Bow," October 18, 1979.

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-302

FLORIDA POWER CORPORATION, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 46 to Facility Operating License No. DPR-72, issued to the 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., and the City of Tallahassee (the licensees) which revised the Technical Specifications (TSs) for operation for the Crystal River Unit No. 3 Nuclear Generating Plant (the facility) located in Citrus County, Florida.

The amendment revises the TSs to authorize Cycle 4 operation of the facility following refueling. It also deletes an improper surveillance requirement for the containment spray additive system.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior

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public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 16, 1981, as supplemented November 20, 1981, (2) Amendment No. 46 to License No. DPR-72, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Crystal River Public Library, 668 N. W. First Avenue, Crystal River, Florida. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 4th day of December 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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