December 14, 1987

Docket No. 50-302

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Mr. W. S. Wilgus
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
P. O. Box 219
Crystal River, Florida 32629

Dear Mr. Wilgus:

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT (TAC NO. 65259)

The Commission has issued the enclosed Amendment No. 103 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 15, 1987, as supplemented July 17, 1987, September 16, 1987 and October 27, 1987.

This amendment provides revised Technical Specifications (TS) to support operation of Crystal River Unit 3 for Fuel Cycle 7.

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

Sincerely,

Original signed by

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 103 to DPR-72
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PD22	PM:PD22	D:PD22	RSB	OGC-Beth
DMiller	HSilver:bd	HBerkow	WHodges	MYoung
12/3/87	12/3/87	12/4/87	12/4/87	12/8/87

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

December 14, 1987

Docket No. 50-302

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Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
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Harley Silver, Project Manager
Project Directorate II-2
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2. Safety Evaluation

cc w/enclosures:
See next page

Mr. W. S. Wilgus
Florida Power Corporation

Crystal River Unit No. 3 Nuclear
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. 103
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 15, 1987, as supplemented July 17, 1987, September 16, 1987 and October 27, 1987, 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.103 , 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

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 14, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 103

FACILITY OPERATING LICENSE NO. DPR-72

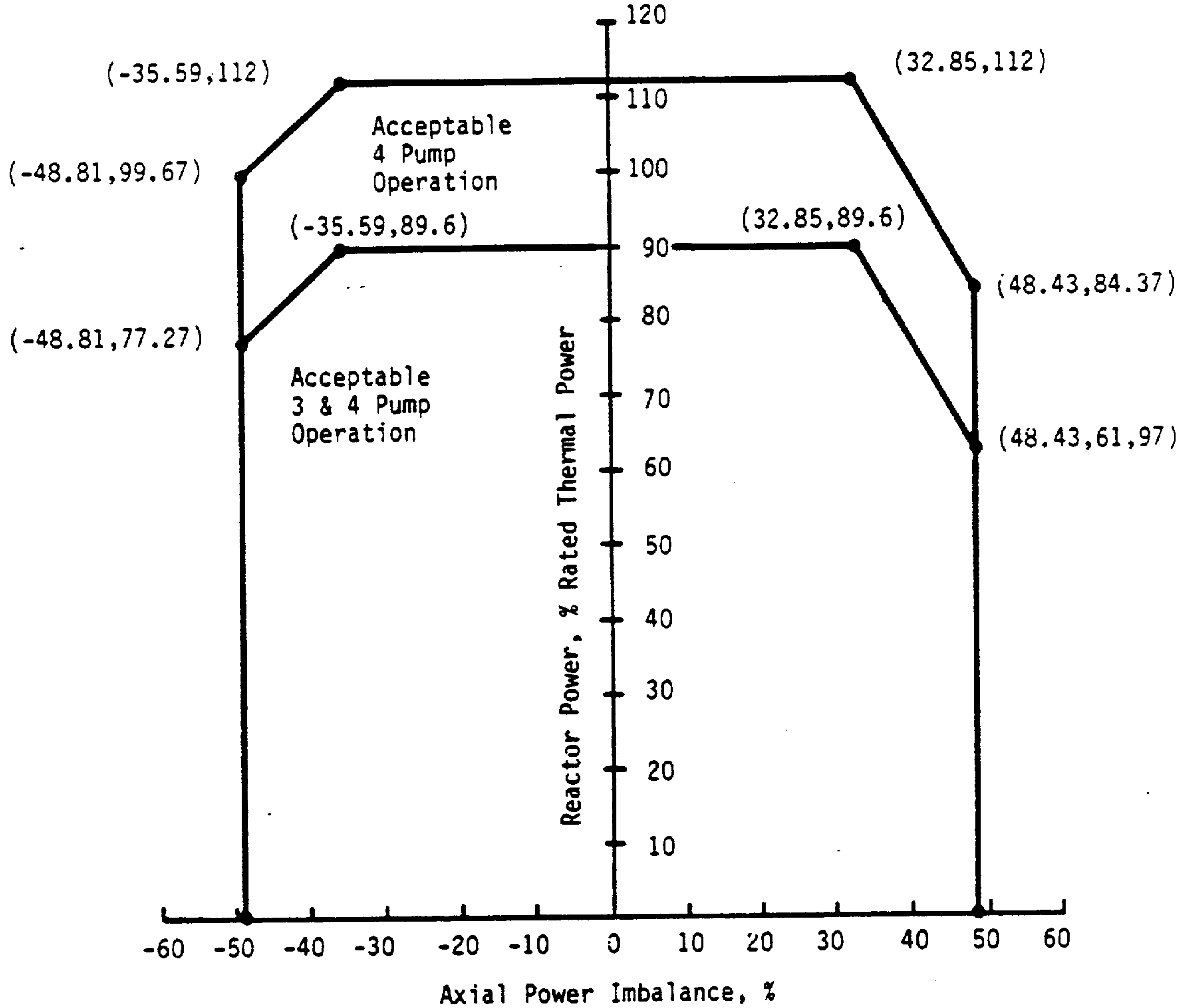
DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached 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.

Remove	Insert
2-3	2-3
2-7	2-7
B2-1	B2-1
B2-2	B2-2
B2-3	B2-3
3/4 1-25	3/4 1-25
3/4 1-27	3/4 1-27
3/4 1-27a	3/4 1-27a
3/4 1-28	3/4 1-28
3/4 1-29	3/4 1-29
3/4 1-29a	3/4 1-29a
3/4 1-30	3/4 1-30
3/4 1-34	3/4 1-34
3/4 1-37	3/4 1-37
3/4 2-1	3/4 2-1
3/4 2-2	3/4 2-2
3/4 2-2a	3/4 2-2a
3/4 2-3	3/4 2-3
3/4 2-3a	3/4 2-3a
3/4 2-11	3/4 2-11
3/4 3-8	3/4 3-8
B3/4 1-2	B3/4 1-2
B3/4 2-1	B3/4 2-1
B3/4 2-2	B3/4 2-2
B3/4 2-3	B3/4 2-3
5-4	5-4

Figure 2.1-2

Reactor Core Safety Limits



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.

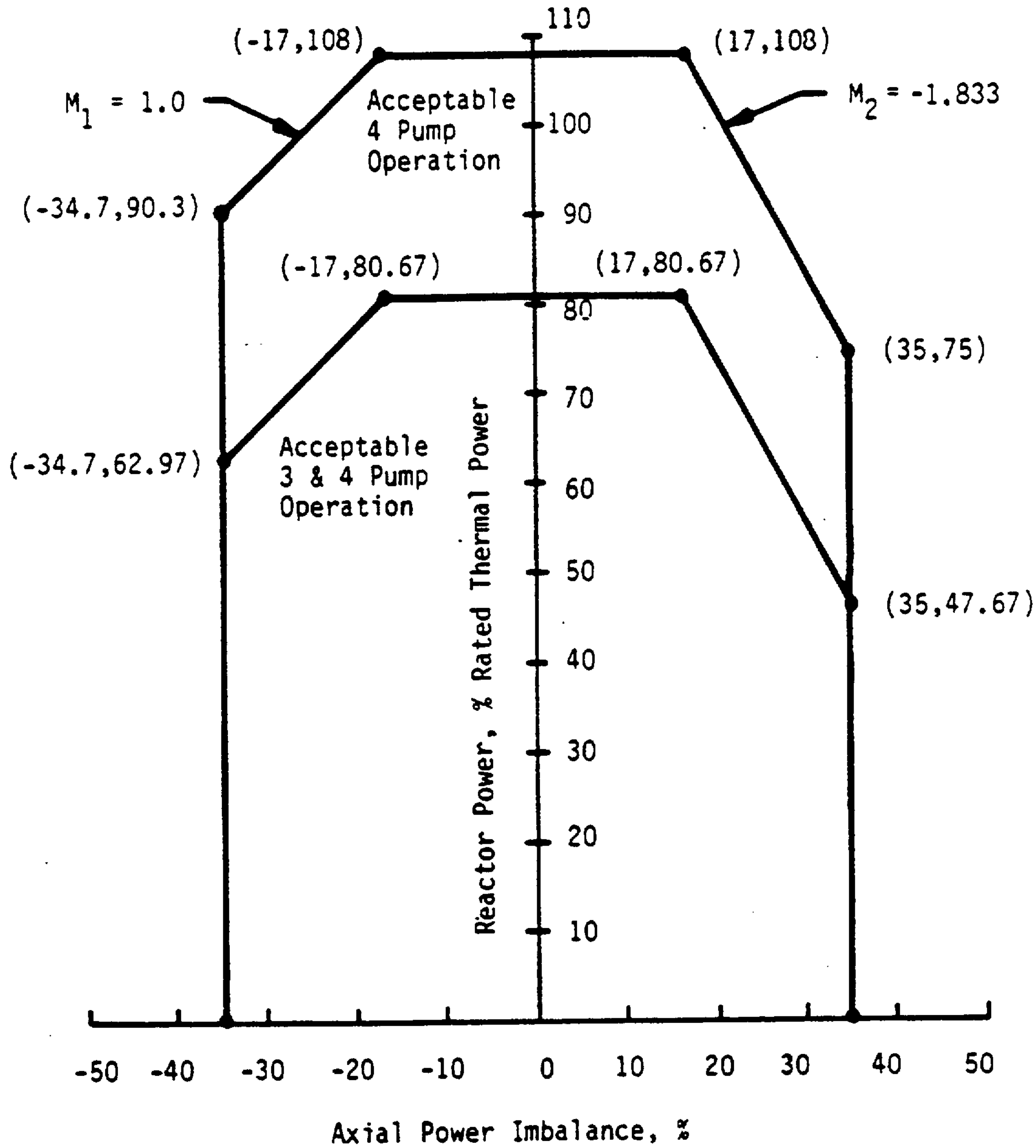
APPLICABILITY: As shown for each channel in Table 3.3-1.

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 but THERMAL POWER and Reactor Coolant Temperature and Pressure can be related to DNB using a Critical Heat Flux (CHF) correlation. 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 B&W-2 and BWC CHF correlations have been developed to predict DNB for axially uniform and non-uniform heat flux distributions. The B&W-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.30 (B&W-2) and 1.18 (BWC). A DNBR of 1.30 (B&W-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur.

The curve presented in Figure 2.1-1 represents the conditions at which the minimum allowable DNBR or greater is predicted for the thermal power and number of 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 consider the effects of potential fuel densification and potential fuel rod bow:

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

2. The combination of radial and axial peak that causes central fuel melting at the hot spot.

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 the DNBR limit 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.

For each curve of BASES Figure 2.1, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (B&W-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22% (B&W-2) or 26% (BWC) for that particular reactor coolant pump situation. The curve for three pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the three pump curve will be above and to the left of the other curves.

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

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

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

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

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

Manual Reactor Trip

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

Nuclear Overpower

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

During normal station operation, reactor trip is initiated when the reactor power level reaches 104.9% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112% which was used in the safety analysis.

REGULATING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

3.1.3 6 The regulating rod groups shall be limited in physical insertion to the acceptable operation region as shown on Figures 3.1-1, 3.1-2, 3.1-3, 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:

a. With the regulating rod groups inserted in the unacceptable operation region, immediately initiate and continue boration at greater than or equal to 10 GPM of 11,600 PPM boric acid solution or its equivalent, until out of the unacceptable operation region.

   Additionally, either:

   1. Restore the regulating groups to within the acceptable region limits within 2 hours of exceeding the acceptable operation region, or

   2. 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 of exceeding the acceptable operation region, or

   3. Be in at least HOT STANDBY within 6 hours of exceeding the acceptable operation region.

b. With the regulating rod groups inserted in the restricted operation region or with any group sequence or overlap outside the specified limits, except for surveillance testing pursuant to Specification 4.1.2.1.2, either:

   1. Restore the regulating groups to within the acceptable region limits within 2 hours of exceeding the acceptable operation region, or

   2. 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 of exceeding the acceptable operation region, or

   3. Be in at least HOT STANDBY within 6 hours of exceeding the acceptable operation region.

---

*See Special Test Exceptions 3.10.1 and 3.10.2.

With K_{eff} greater than or equal to 1.0.

SURVEILLANCE REQUIREMENTS

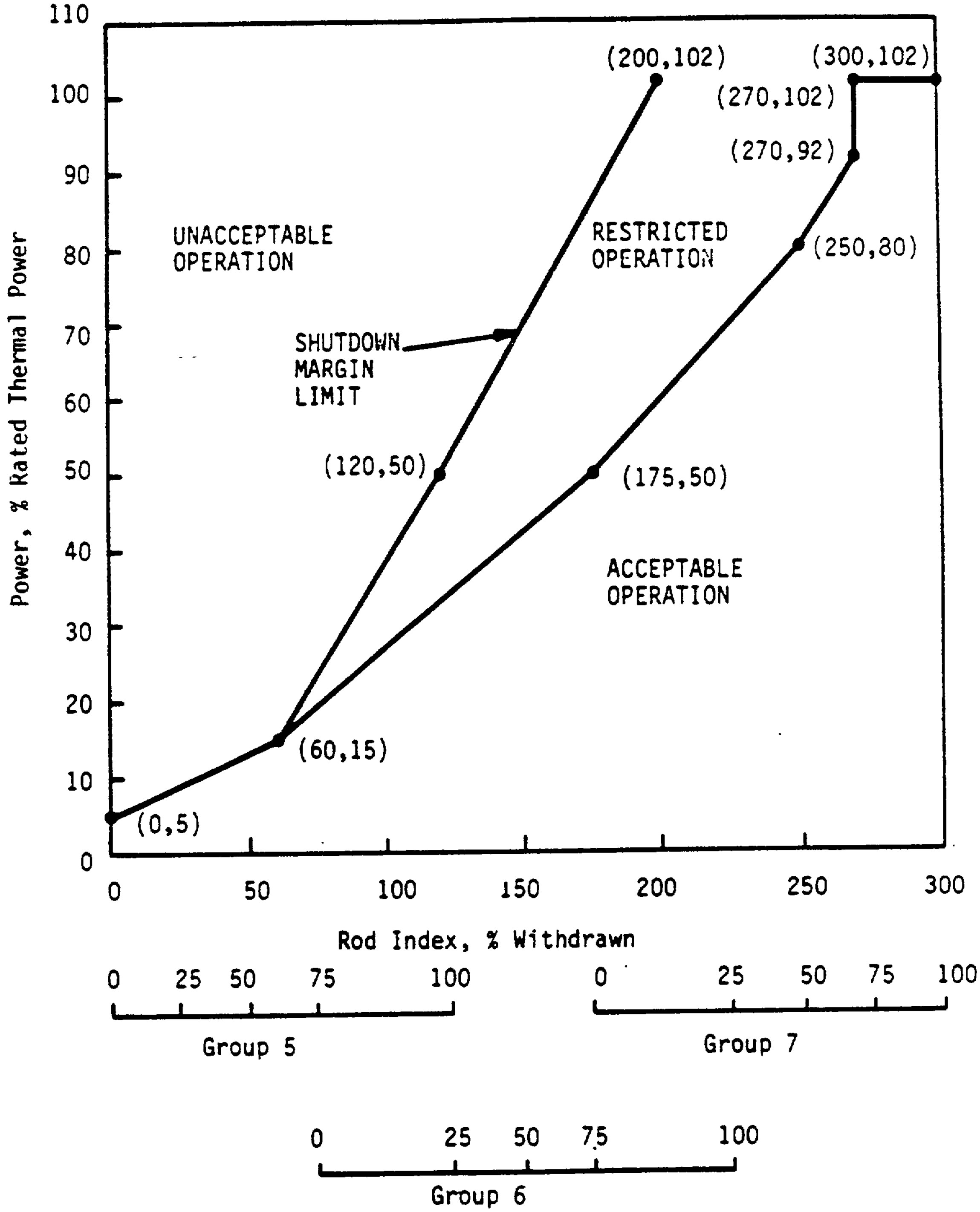
4.1.3.6 The position of each regulating group shall be determined to be within the insertion, sequence and overlap limits at least once every 12 hours except when:

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 to 500±10 EFPD



Note: This Figure shall be used up to complete APSR withdrawal per Specification 3.1.3.9

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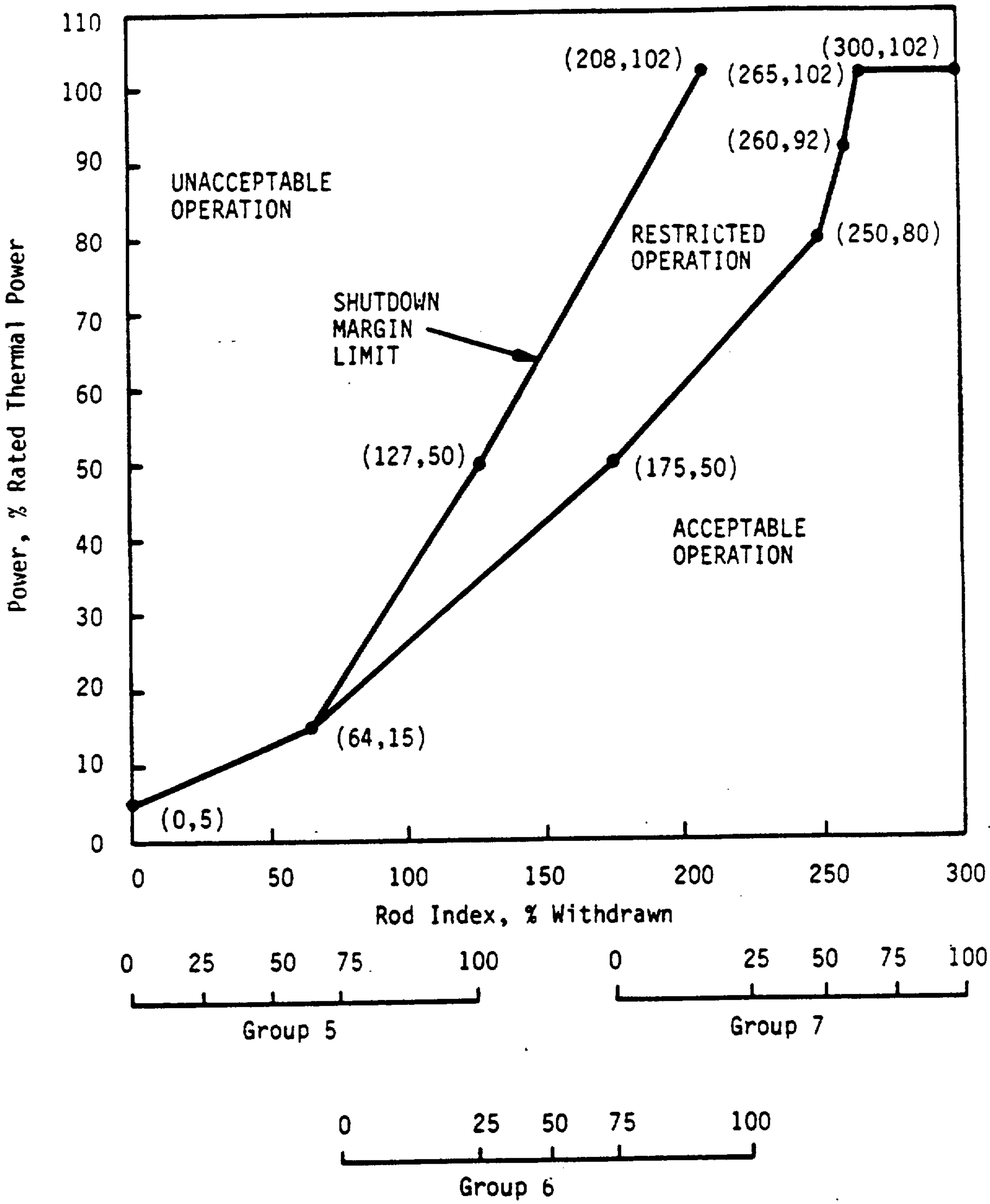
Figure 3.1-2

Regulating Rod Group Insertion Limits for Four-Pump Operation After 500±10 EFPD

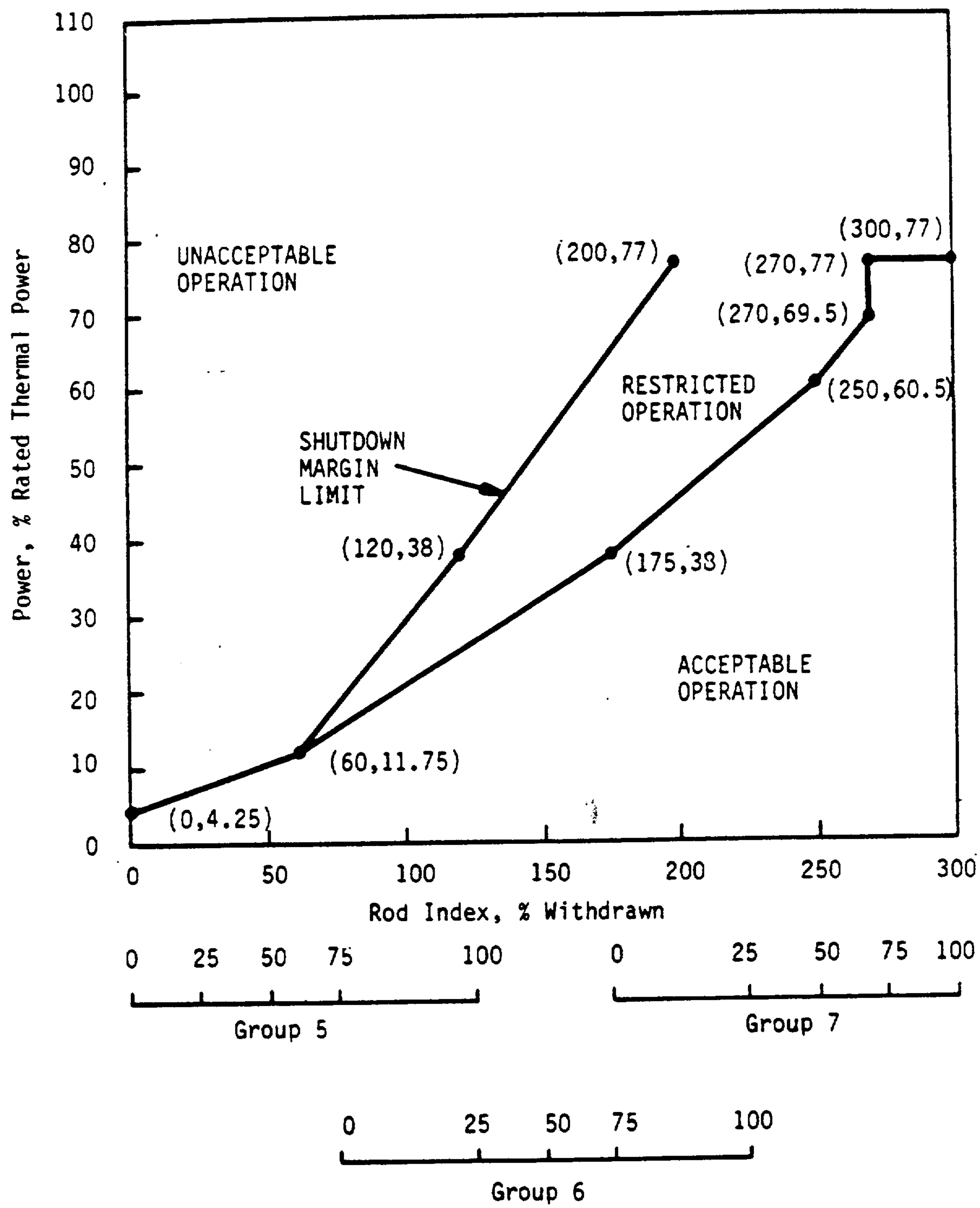


Note: This Figure shall be used after complete APSR withdrawal per Specification 3.1.3.9

-DELETED-

Figure 3.1-3

Regulating Rod Group Insertion Limits for Three-Pump Operation From 0 to 500±10 EFPD

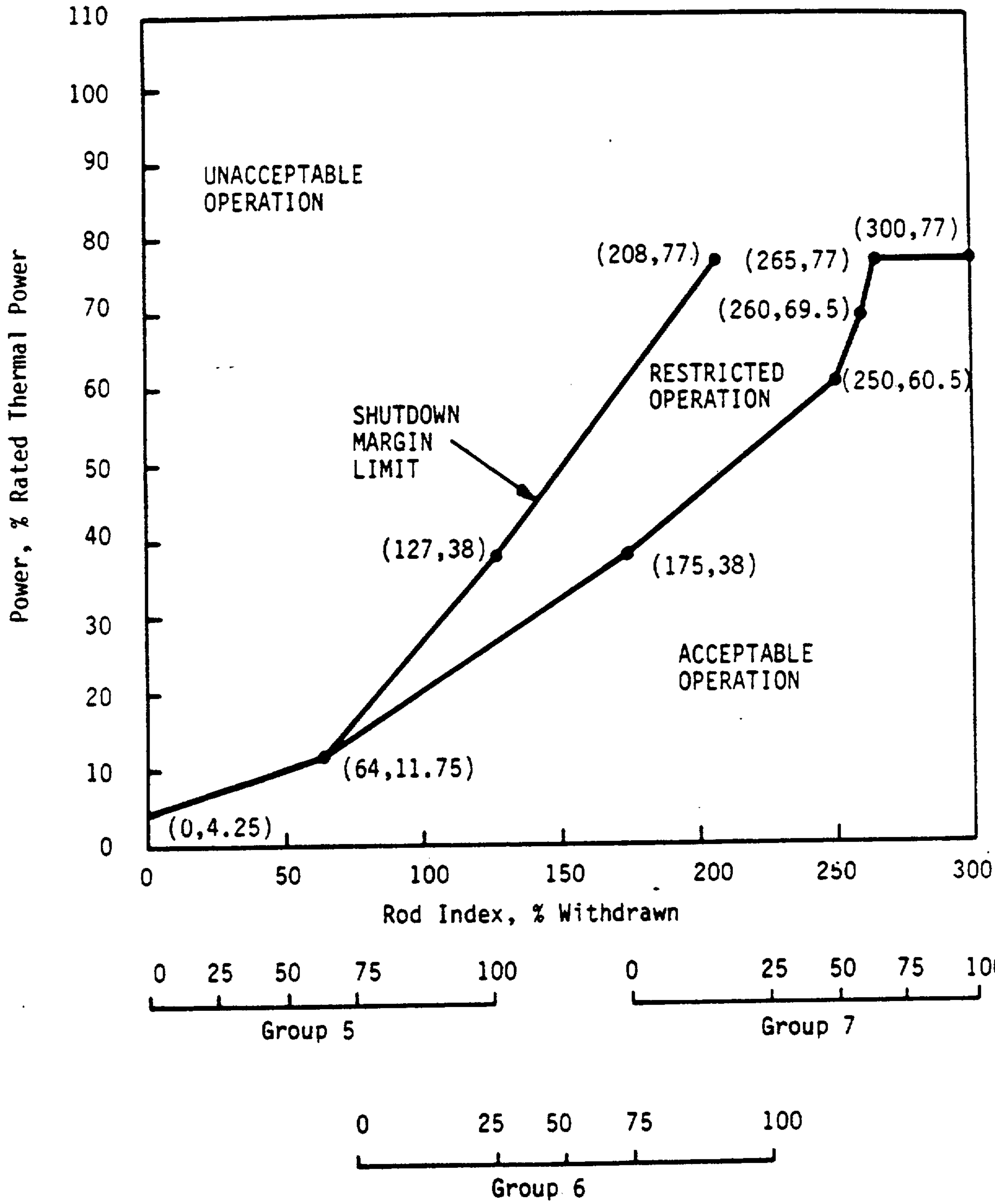


Note: **This Figure shall be used up to complete APSR withdrawal per Specification 3.1.3.9**

-DELETED-

Figure 3.1-4

Regulating Rod Group Insertion Limits for Three-Pump Operation After 500±10 EFPD



Note: This Figure shall be used after complete APSR withdrawal per Specification 3.1.3.9

-DELETED-

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 oup Designations
for Crystal River 3 Cycle 7

X — Fuel Transfer Canal →

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
A															
B						1		6		1					
C					3		5		5		3				
D				7		8		7		8		7			
E			3		5		4		4		5		3		
F		1		8		6		2		6		8		1	
G			5		4		2		2		4		5		
H (W — Y)		6		7		2				2		7		6	
K			5		4		2		2		4		5		
L		1		8		6		2		6		8		1	
M			3		5		4		4		5		3		
N				7		8		7		8		7			
O					3		5		5		3				
P						1		6		1					
R															

Z

$\boxed{X}$ Group Number

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 490 EFPD, the APSR's may be positioned as necessary. The APSR's shall be completely withdrawn (100%) by 510 EFPD. Between 490 and 510 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-

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

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

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

ACTION:

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

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

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

SURVEILLANCE REQUIREMENTS

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

---

* See Special Test Exception 3.10.1.

Figure 3.2-1

Axial Power Imbalance Envelope for Four-Pump Operation From 0 EFPD to EOC

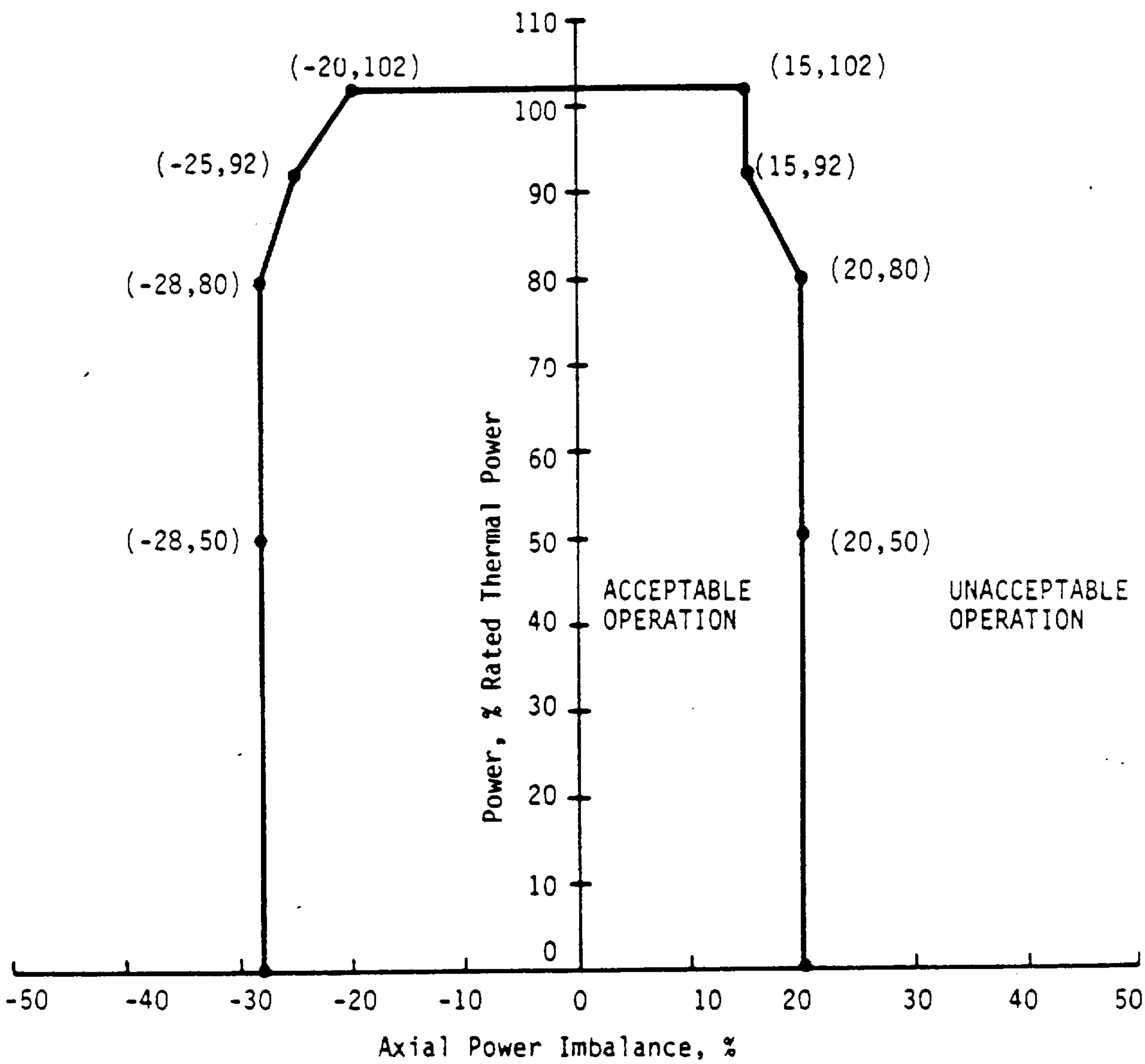


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Figure 3.2-2

Axial Power Imbalance Envelope for Three-Pump Operation From 0 EFPD to EOC

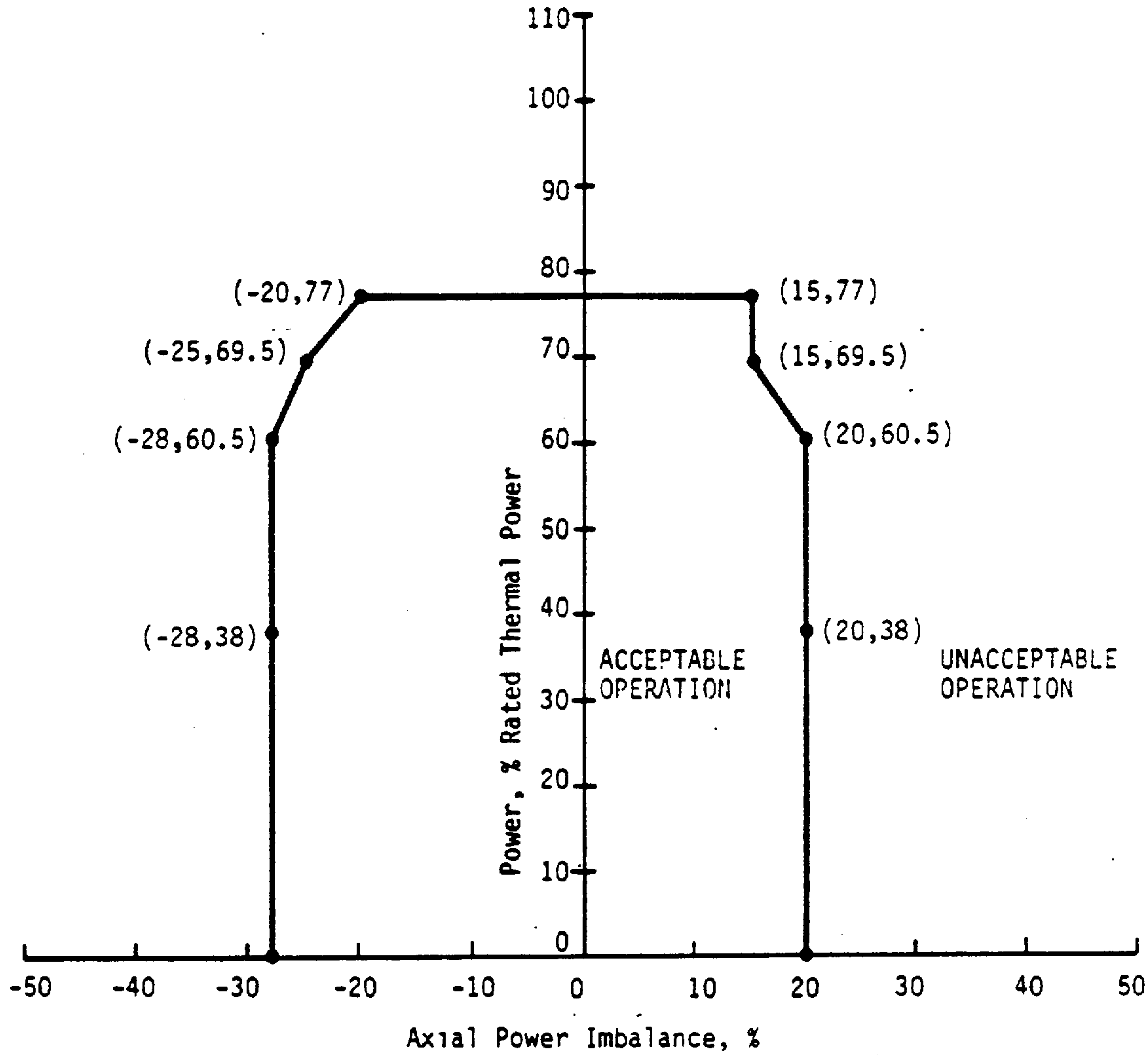

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TABLE 3.2-2

QUADRANT POWER TILT LIMITS

	STEADY STATE LIMIT	TRANSIENT LIMIT	MAXIMUM LIMIT
QUADRANT POWER TILT as Measured by:			
Symmetrical Incore Detector System	4.12	10.03	20.0
Power Range Channels	1.96	6.96	20.0
Minimum Incore Detector System	1.9	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.

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
15.	Anticipatory Reactor Trip - Main Turbine	S	R	M	1
16	Anticipatory Reactor Trip - Both Main Feedwater Pumps	S	R	M	1

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_1) as follows:

$$\frac{RTP}{TP} \quad (API_O - API_1) = \text{Imbalance Error}$$

Recalibrate if the absolute value of the Imbalance Error is equal to or greater than 2.5%.

(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.

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 atleast 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.

BASES

3/4.1.1.4 MINIMUM TEMPERATURE FOR CRITICALITY

This specification ensures that the reactor will not be made critical with the Reactor Coolant System average temperature less than 525°F. This limitation is required to ensure that (1) the moderator temperature coefficient is within its analyzed temperature range, (2) the protective instrumentation is within its normal operating range, (3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and (4) the reactor pressure vessel is above its minimum RTNDT temperature.

3/4.1.2 BORATION SYSTEMS

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

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

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs from full power equilibrium xenon conditions and requires either 5,400 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 45,000 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% $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 490 gallons of 11,600 ppm boron from the boric acid storage system or 2,502 gallons of 2,270 ppm boron from the borated water storage tank. To envelope future cycle BWST and BAST contained borated water volume requirements, a minimum volume of 13,500 gallons and 600 gallons, respectively, are specified.

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

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 allowable fuel melt limit locally, and from going below minimum allowable DNBR 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 4.92, 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 within the limit 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.

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 4.0 (nominal) weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 60 safety and regulating (including extended life control rods) and 8 axial power shaping (APSR) control rods. Except for the extended life control rods, the safety and regulating control rods shall contain a nominal 134 inches of absorber material. The extended life control rods shall contain a nominal 139 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium, and 5 percent cadmium. Except for the extended life control rods, all control rods shall be clad with stainless steel tubing. The extended life control rods shall be clad with Inconel. 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. 103 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 April 15, 1987 (Ref. 1), Florida Power Corporation (the licensee) submitted an application to reload Crystal River Unit No. 3 and operate it for Cycle 7. To support the application, the licensee submitted report BAW-1988 (Ref. 2) entitled, "Crystal River Unit 3 Cycle 7 Reload Report," and proposed changes to the Unit 3 Technical Specifications. The licensee submitted supplemental information on July 17, 1987 (Ref. 9) and September 16, 1987 (Ref. 9A). In addition, supplemental information was submitted on October 27, 1987 (Ref. 2A) to reflect an expansion of the end of Cycle 6 shutdown window to 465-61/+10 effective full power days (EFPD). The actual Cycle 6 length was approximately 412 EFPD.

The Cycle 7 core consists of 177 fuel assemblies, each of which is a 15 by 15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 7 is to have an operating length of approximately 550 EFPD. As has been the case for Cycle 6, Cycle 7 will be operated in a rods out, 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 68 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 remains at 2568 MWt.

2.0 EVALUATION OF THE FUEL SYSTEM DESIGN

Cycle 7 will contain eight Mark B3 fuel assemblies in Batch 2C, 89 Mark B4 assemblies in Batches 4B, 5C, 7C, and 8, and 80 Mark BZ assemblies in Batch 9. All of these fuel assemblies are mechanically interchangeable. The Batch 8 and 9 assemblies contain a redesigned holddown spring made from Inconel 718 which provides added margin over the previous spring design. The Mark BZ design is similar to the Mark B fuel assembly except that the six intermediate Inconel spacer grids have been replaced with Zircaloy grids.

For Cycle 7, a significant portion of the core will contain Mark BZ fuel. The design (Ref. 3) of these assemblies has been reviewed and approved by the NRC (Ref. 4). However, the NRC safety evaluation stated that a licensee incorporating this design was required to submit a plant-specific analysis of combined seismic and loss-of-coolant accident (LOCA) loads according to Appendix A of Standard Review Plan 4.2 (Ref. 5). The staff concludes that the analysis which was presented in the Rancho Seco Cycle 7 reload report (Ref. 3) envelopes the Crystal River 3 plant design requirements and, therefore, the margin of safety reported for the Mark BZ fuel is applicable to Crystal River 3. Therefore, the Mark BZ assemblies are acceptable for use in Cycle 7.

Although the prepressure of the Batch 9 fuel assemblies has been lowered by 50 psi in order to provide a higher burnup limit for pin pressure, the Batch 2C Mark B3 assemblies have the lowest prepressure. Because of the lower prepressure and the previous incore exposure of Batch 2C fuel, it is the most limiting in terms of cladding creep collapse. The licensee has stated that the cladding collapse time for the most limiting Cycle 7 assembly was conservatively determined to be greater than the maximum projected residence time for any Cycle 7 assembly. The methods and procedures used for the analyses (Ref. 6) have been previously reviewed and approved by the staff. The staff concludes that cladding collapse has been appropriately considered and will not occur for Cycle 7 operation.

The cladding stress and strain analyses for the Cycle 7 fuel designs are either bounded by conditions previously analyzed for Crystal River 3 or were analyzed specifically for Cycle 7 using methods and limits previously reviewed and approved by the NRC. The staff concludes that the analysis of cladding stress and strain has been appropriately considered for Cycle 7 operation and is acceptable.

The thermal behavior of all fuel in the Cycle 7 core is virtually identical. The thermal analysis was performed with the approved TACO2 code (Ref. 7) and the Cycle 7 core protection limits were based on the calculated linear heat rate (LHR) to centerline fuel melt limits. These limiting values are satisfactorily incorporated into the Technical Specifications for Cycle 7 through the operating limits on rod index and axial power imbalance.

Standard Review Plan 4.2, Section II.A.1(f) contains the requirement that the fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Based on TACO2 analyses, the licensee has stated that the internal pressure in the highest burnup rod of each fuel type will not reach the nominal reactor coolant system (RCS) pressure of 2200 psia. The staff finds this acceptable and concludes that the fuel rod internal pressure limits have been adequately considered for Cycle 7 operation.

Subsequent to the staff's initial review of the Cycle 7 reload report, Unit 3 became limited to three pump operation and a reevaluation of the end-of-cycle (EOC) 6 shutdown window from 465 EFPD to as low as 404 EFPD was made by the licensee. A reduced Cycle 6 length (404 EFPD) results in lower burnup for the limiting fuel in Cycle 7 with respect to internal rod pressure, cladding creep collapse, and cladding strain. The power in the limiting fuel is no greater than that based on a Cycle 6 length of 465 EFPD. Therefore, the Cycle 7 fuel design analyses evaluated for a Cycle 6 length of 465 EFPD remain acceptable.

3.0 EVALUATION OF NUCLEAR DESIGN

The nuclear design parameters characterizing the Crystal River 3 Cycle 7 core have been computed by methods previously used and approved for B&W reactors (Ref. 8). Comparisons have been made between the parameters for Cycle 6 and Cycle 7. Changes in the cycle length, feed batch size and enrichment, burnable poison rod assembly (BPRA) loading, and shuffle pattern between cycles accounts for the differences in control rod worths, critical boron concentrations, and moderator temperature coefficients. In addition, eight extended life control rod assemblies similar to those placed in service at Arkansas Nuclear One, Unit 1 for Cycle 7 will be used in Cycle 7 for Crystal River (Ref. 9A).

The fresh Batch 9 Mark BZ fuel will have an initial enrichment of 3.84 weight percent U-235. The staff finds this acceptable since the Crystal River 3 spent fuel pool has been designed to store fuel with a maximum enrichment of 4.0 weight percent U-235.

Shutdown margin calculations for Cycle 7 include the effects of poison material depletion, a 10% calculational uncertainty, allowance for rod bite and neutron flux redistribution, as well as a maximum worth stuck rod. Beginning-of-cycle (BOC), middle-of-cycle (500 EFPD), and end-of-cycle (EOC) shutdown margins show adequate reactivity worth exists above the total required worth during the cycle. Shutdown margins at BOC and EOC are 3.81% delta k/k and 2.66% delta k/k, respectively, compared to the minimum required value of 1.0% delta k/k.

The effect of the reduced Cycle 6 length of 404 EFPD on the nuclear analysis for Cycle 7 was evaluated by the licensee. The Cycle 7 reactivity parameters remain bounded by the values used in the referenced safety analyses.

Based on its review, the staff concludes that approved methods have been used, that the nuclear design parameters meet applicable criteria and that the nuclear design of Crystal River 3 is acceptable.

4.0 EVALUATION OF THERMAL HYDRAULIC DESIGN

Although a full Mark BZ core and a full Mark B core provide practically the same departure from nucleate boiling (DNB) margin for both steady-state and transient conditions (Ref. 4), incompatibility in the hydraulic characteristics has an effect on thermal margin during transitional mixed core cycles when both Mark BZ and Mark B fuel assemblies co-exist in the core. Since the Mark BZ assemblies have a higher hydraulic resistance due to the BPRA retainers and the Zircaloy intermediate spacer grids, some of the coolant flow is diverted from the Mark BZ fuel to the lower-powered Mark B fuel. The fact that the Mark BZ assemblies have less flow in a mixed core results in lower maximum allowable power peaking and a lower enthalpy rise factor required in order to maintain the same DNBR limit compared to a whole core of Mark BZ fuel. The staff requested additional information justifying that the increased bypass flow of 8.8% used in the thermal-hydraulic design evaluation for the full Mark BZ core provides sufficient margin to offset this core penalty required for the Cycle 7 mixed core of Mark BZ and Mark B fuel assemblies. In response (Ref. 9), the licensee described analyses performed for a mixed core with a core bypass flow fraction of 7.8% and a full Mark BZ core with an 8.8% bypass flow. The results indicate that a full Mark BZ core with an 8.8% bypass flow conservatively models the mixed core of Cycle 7.

A B&W topical report (Ref. 10) discussing the mechanisms and resulting effects of bowing in B&W fuel has been reviewed and approved (Ref. 11). The report concludes that the DNBR penalty due to rod bow need not be imposed for those assemblies with significant burnup because the power production capability of the fuel decreases sufficiently with irradiation to offset the effects of bowing. Post-irradiation measurements on Mark BZ lead demonstration assemblies verified that the methodology of Ref. 10 conservatively predicts the rod bow in Mark BZ assemblies. Therefore, the staff concludes that no rod bow penalty need be considered for Cycle 7 operation.

5.0 ACCIDENT AND TRANSIENT ANALYSES

The important physics, thermal-hydraulic, and kinetics parameters for Cycle 7 have been compared to the values used in the FSAR (Ref. 12) and/or the fuel densification report (Ref. 13). The licensee has shown that the Cycle 7 values are bounded by those previously used. The licensee has also determined that, except for the boron dilution event, the initial conditions of the transients and accidents in Cycle 7 are bounded by either the FSAR, the fuel densification report, or previous reload analyses. These analyses have been previously accepted by the NRC.

The boron dilution event was reevaluated for Cycle 7 because of the longer cycle length, different core kinetics parameters, and higher BOC boron concentration. The resulting thermal power, RCS pressure, and subcriticality margin are within the current acceptance criteria for the boron dilution event, and therefore, the results are acceptable for Cycle 7.

B&W has performed a generic LOCA analysis for the B&W 177-FA lowered-loop nuclear steam supply system (NSSS) using the final acceptance criteria (FAC) emergency core cooling system (ECCS) evaluation model (Ref. 14). The combination of average fuel temperature as a function of LHR and the lifetime pin pressure data used is conservative relative to those calculated for this cycle. Three sets of bounding values for allowable LOCA peak LHRs for Cycle 7 are given as a function of core height. These limits apply during the periods 0 to 30 EFPD, 30 to 75 EFPD, and for the balance of the cycle. These results are based upon a bounding analytical assessment of NUREG-0630 on LOCA and operating LHR limits performed by B&W (Ref. 15). The B&W analyses have been approved by the NRC staff and the LHR limits are satisfactorily incorporated into the Technical Specifications for Cycle 7 through the operating limits on control rod withdrawal index and axial power imbalance.

Because of the increase in the fraction of fissions produced by plutonium resulting from changes in fuel management techniques, the radiological dose consequences of the accidents presented in the FSAR were reevaluated for Cycle 7. A comparison of the doses presented in the FSAR to those calculated for Cycle 7 indicates that the Cycle 7 values are either bounded by the FSAR values or are a small fraction of the 10 CFR 100 limits, i.e., below 30 rem to the thyroid and 2.5 rem to the whole body. Therefore, the staff finds the radiological impact of accidents during Cycle 7 meets all safety criteria and is acceptable.

6.0 TECHNICAL SPECIFICATION CHANGES

Crystal River Unit 3 Cycle 7 Technical Specifications have been modified to support a longer fuel cycle length (550 days) as well as various operational and design changes. These changes include reactor safety limit and overpower trip limits, rod insertion limits, extended life control rod description, control rod locations, axial power imbalance limits and imbalance error and quadrant power tilt limits. The staff has reviewed the proposed changes (Refs. 1, 9, and 9A) and finds them acceptable because they are derived from analyses performed using approved methods and have been appropriately considered in the safety analyses. In addition, an evaluation of the effects of a reduction in Cycle 6 length to 404 EFPD resulted in no revisions to the Cycle 7 Technical Specifications, which were based on a Cycle 6 length of 465 EFPD.

7.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, thermal-hydraulic, and accident information presented in the Crystal River 3 Cycle 7 reload report and supplementary information and finds the proposed reload and the associated modified Technical Specifications acceptable.

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.

10. REFERENCES

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13. "Fuel Densification Report," BAW-1397, Babcock & Wilcox, August 1973.

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Dated: December 14, 1987

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