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	RTedesco	ACRS-16
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Docket No. 50-302

Mr. J. A. Hancock
Director, Nuclear Operations
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Hancock:

The Commission has issued the enclosed Amendment No. 32 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your submittal dated March 21, 1980, as revised and supplemented April 14 and 30, June 6 and 13, and July 22 and 31, 1980.

This amendment revises the Appendix A Technical Specifications to permit power operation during Cycle 3 at the previously authorized power level of 2452 MWt. Some portions of your proposed Technical Specifications in support of operation with the reloaded core have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff.

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

Sincerely,

Original signed by
Robert W. Reid

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

Enclosures:
1. Amendment No. 32
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

6D

8008140483 P

*See previous yellow for concurrences

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	AD-OR:DL	OELD	
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Robert W. Reid, Chief
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cc w/enclosures: See next page

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	~~AD-OR:DL~~	OELD	
SURNAME	RIngram	MFairtile/cb	~~RReid~~	TNovak	KARMAN	
DATE	7/10/80	7/17/80	7/17/80	7/17/80	7/18/80	

NRC FORM 318 (9-76) NRCM 0240 ☆U.S. GOVERNMENT PRINTING OFFICE: 1979-289-369

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
August 4, 1980

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Docket No. 50-302

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: CRYSTAL RIVER UNIT NO. 3

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment No. 32
  Referenced documents have been provided PDR

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#4:DL					
SURNAME→	RIngram/cb					
DATE→	8/4/80					

NRC FORM 102 (1-76)

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

August 1, 1980

Docket No. 50-302

Mr. J. A. Hancock
Director, Nuclear Operations
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Hancock:

The Commission has issued the enclosed Amendment No. 32 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your submittal dated March 21, 1980, as revised and supplemented April 14 and 30, June 6 and 13, and July 22 and 31, 1980.

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Sincerely,

Robert W. Reid

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

Enclosures:
1. Amendment No. 32
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

8008140485

Crystal River-3 50-302
Florida Power Corporation

cc w/enclosure(s):
Mr. S. A. Brandimore
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

U. S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

Crystal River Public Library
Crystal River, Florida 32629

Mr. J. Shreve
The Public Counsel
Room 4 Holland Bldg.
Tallahassee, Florida 32304

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
Suite 420, 7735 Old Georgetown Road
Bethesda, Maryland 20014

cc w/enclosures & incoming
dtd: 3/21/80, 4/14&30/80, & 6/6&13/80, 7/22 & 8/1/80
Bureau of Intergovernmental Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

Mr. Tom Stetka, Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 2082
Crystal River, Florida 32629

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.32
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al (the licensees) dated March 21, 1980, as revised and supplemented April 14 and 30, June 6 and 13, and July 22 and 31, 1980, 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:

8008140485

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 32 , are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: August 1, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 32

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.

Remove Pages	Insert Pages
III	III
2-3	2-3
2-5	2-5
2-7	2-7
B 2-1	B 2-1
B 2-5	B 2-5
B 2-6	B 2-6
B 2-8	B 2-8
3/4 1-1	3/4 1-1
3/4 1-2a	3/4 1-2a
3/4 1-2b	
3/4 1-7	3/4 1-7
3/4 1-8	3/4 1-8
3/4 1-10	3/4 1-10
	3/4 1-10a
3/4 1-13	3/4 1-13
3/4 1-14	3/4 1-14
3/4 1-16	3/4 1-16
3/4 1-16a	
3/4 1-17	3/4 1-17
3/4 1-27	3/4 1-27
3/4 1-28	3/4 1-28
3/4 1-29	3/4 1-29
3/4 1-30	3/4 1-30
3/4 1-38	3/4 1-38
3/4 1-39	3/4 1-39
3/4 2-2	3/4 2-2
3/4 2-3	3/4 2-3
3/4 2-11	3/4 2-11
3/4 2-13	3/4 2-13
3/4 4-1	3/4 4-1
3/4 10-4	3/4 10-4
B 3/4 1-1	B 3/4 1-1
B 3/4 1-2	B 3/4 1-2
B 3/4 1-3	B 3/4 1-3

INDEX

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

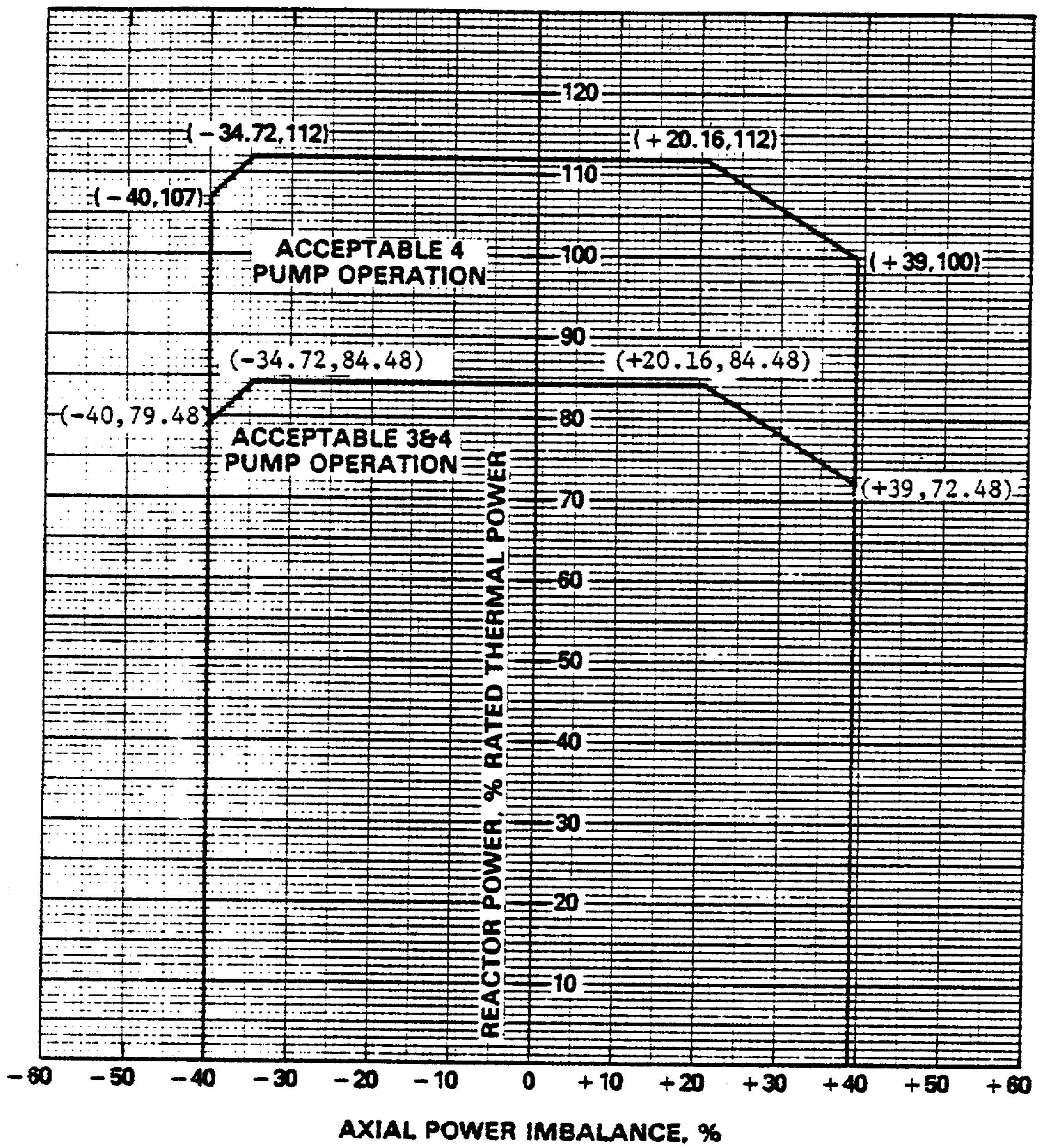


FIGURE 2.1-2

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.

TABLE 2.2-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Nuclear Overpower	≤ 105.5% of RATED THERMAL POWER with four pumps operating ≤ 77.98% of RATED THERMAL POWER with three pumps operating	≤ 105.5% of RATED THERMAL POWER with four pumps operating ≤ 77.98% of RATED THERMAL POWER with three pumps operating
3. RCS Outlet Temperature-High	≤ 619°F	≤ 619°F
4. Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE (1)	Trip Setpoint not to exceed the limit line of Figure 2.2-1.	Allowable Values not to exceed the limit line of Figure 2.2-1.
5. RCS Pressure-Low(1)	≥ 1800 psig	≥ 1800 psig
6. RCS Pressure-High	≤ 2300 psig	≤ 2300 psig
7. RCS Pressure-Variable Low(1)	≥ (11.80 T_{out} °F - 5209.2) psig	≥ (11.80 T_{out} °F - 5209.2) psig

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTION UNIT	TRIP SETPOINT	ALLOWABLE VALUES
8. Reactor Containment Vessel Pressure High	$\leq$ 4 psig	$\leq$ 4 psig

(1) Trip may be manually bypassed when RCS pressure $\leq$ 1720 psig by actuating Shutdown Bypass provided that:

a. The Nuclear Overpower Trip Setpoint is < 5% of RATED THERMAL POWER
b. The Shutdown Bypass RCS Pressure - High Trip Setpoint of $\leq$ 1720 psig is imposed, and
c. The Shutdown Bypass is removed when RCS Pressure > 1800 psig.

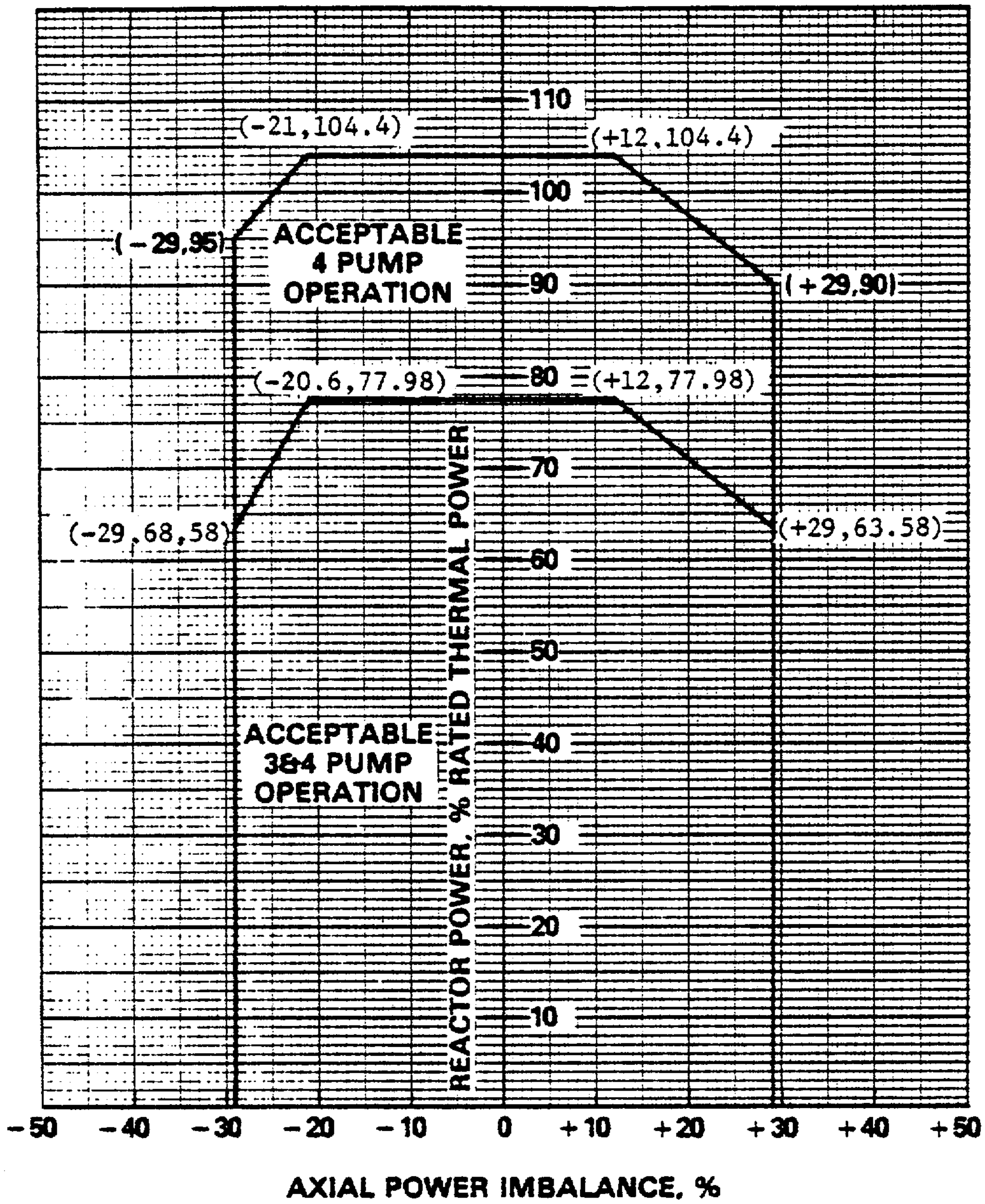


FIGURE 2.2-1

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

2.1 SAFETY LIMITS

BASES

2.1.1 and 2.1.2 REACTOR CORE

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

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

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

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

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

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

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

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

1. The 1.30 DNBR limit produced by a nuclear power peaking factor of $F_Q^N = 2.57$ or the combination of the radial peak, axial peak and position of the axial peak that yields no less than a 1.30 DNBR.

2. The combination of radial and axial peak that causes central fuel melting at the hot spot. The limit is 19.7 kw/ft.

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

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

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

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

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

RCS Outlet Temperature - High

The RCS Outlet Temperature High trip $\leq$ 619°F prevents the reactor outlet temperature from exceeding the design limits and acts as a backup trip for all power excursion transients.

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

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

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

1. Trip would occur when four reactor coolant pumps are operating if power is $\geq$ 104.4% and reactor flow rate is 100%, or flow rate is $\leq$ 95.78% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is > 77.98% and reactor flow rate is 74.7%, or flow rate is $\leq$71.84% 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 1.044% for a 1% flow reduction.

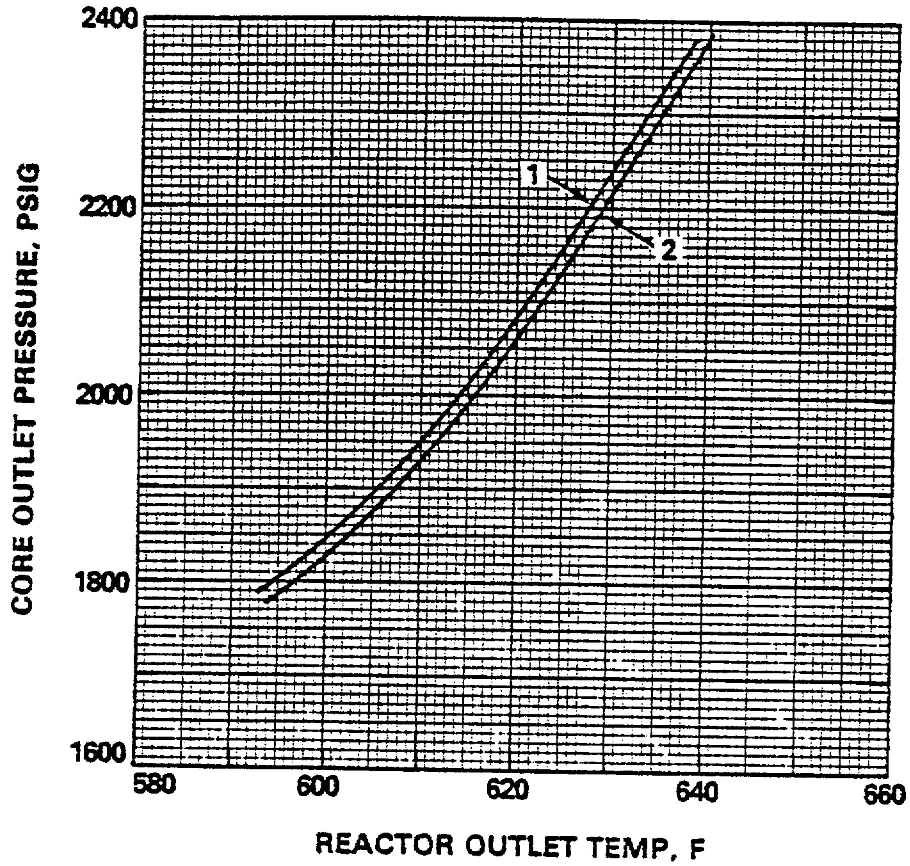
RCS Pressure - Low, High and Variable Low

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, (11.80 T_{out} °F-5209.2) 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 (11.80 T_{out} °F-5249.2) psig.



REACTOR COOLANT FLOW

CURVE	FLOW (% DESIGN)	POWER (RTP)	PUMPS OPERATING (TYPE OF LIMIT)
1	139.7×10^6 (106.5%)	117.3%	4 PUMPS (DNBR)
2	104.4×10^6 (79.6%)	90.5%	3 PUMPS (DNBR)

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM ALLOWABLE POWER FOR MINIMUM DNBR

BASES FIGURE 2.1

LIMITING SAFETY SYSTEM SETTINGS

BASES

Reactor Containment Vessel Pressure - High

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

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 1% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, and 3.

ACTION:

With the SHUTDOWN MARGIN < 1% $\Delta k/k$, 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

4.1.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 1% $\Delta k/k$:

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

b. When in MODES 1 or 2#, at least once per 12 hours, by verifying that regulating rod groups withdrawal is within the limits of Specification 3.1.3.6.

c. When in MODE 2## within 4 hours prior to achieving reactor criticality by verifying that the predicted critical control rod position is within the limits of Specification 3.1.3.6.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading by consideration of the factors of e. below, with the regulating rod groups at the maximum insertion limit of Specification 3.1.3.6.

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

##With $K_{eff} < 1.0$.

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODE 3, at least once per 24 hours by consideration of the following factors:

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

4.1.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm 1\% \Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.1.e above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

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$3.0% $\Delta k/k$.

APPLICABILITY: MODES 4 and 5.

ACTION:

MODE 4

With the SHUTDOWN MARGIN <3.0% $\Delta k/k$, 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.

MODE 5

With the SHUTDOWN MARGIN <3.0% $\Delta k/k$, 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 $\geq$ 3.0% $\Delta k/k$:

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

b. 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% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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

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 3.0% $\Delta k/k$ at 200°F within the next 6 hours; restore the flow path to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

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 PUMP - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least one makeup pump in the boron injection flow path required by Specification 3.1.2.1 shall be OPERABLE and capable of being powered from an OPERABLE emergency bus.

APPLICABILITY: MODE 5*.

ACTION:

With no makeup pump OPERABLE, suspend all operations involving positive reactivity changes until at least one makeup pump is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

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

*RCS Pressure $\geq$ 150 psig.

MAKEUP PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.4.1 At least two makeup pumps shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

ACTION:

With only one makeup pump OPERABLE, restore at least two makeup pumps 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 two makeup pumps to OPERABLE status within the next 7 days or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

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 3.0% $\Delta k/k$ at 200°F and be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

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

*With RCS pressure $\geq$ 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% Δk/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 3.0% Δk/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 6615 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 2270 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 6615 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 2270 and 2450 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 3.0% $\Delta k/k$ 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 $\pm$ 6.5% (indicated position) of their group average height.

APPLICABILITY: MODES 1* and 2*.

ACTION:

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

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

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

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

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

        a) An analysis of the potential ejected rod worth is performed within 72 hours and the rod worth is determined to be $\leq$ 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.

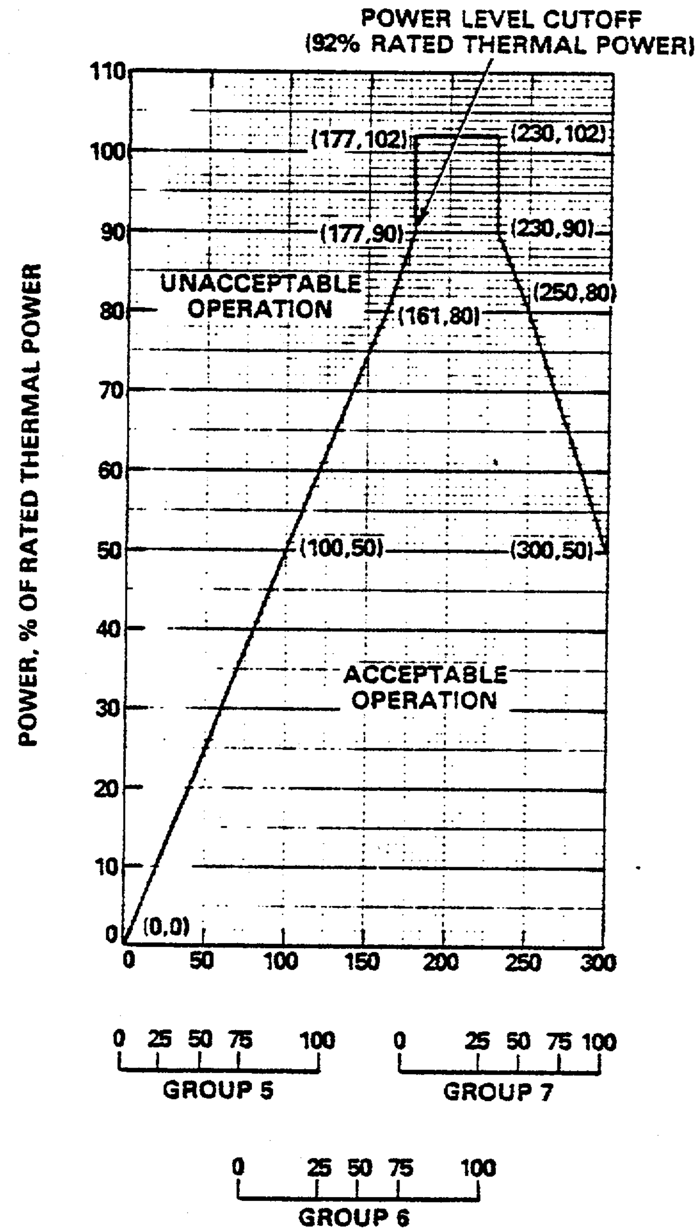


FIGURE 3.1-1

REGULATING ROD GROUP INSERTION LIMITS FOR 4 PUMP OPERATION FROM 0 EFPD TO 250 ± 10 EFPD

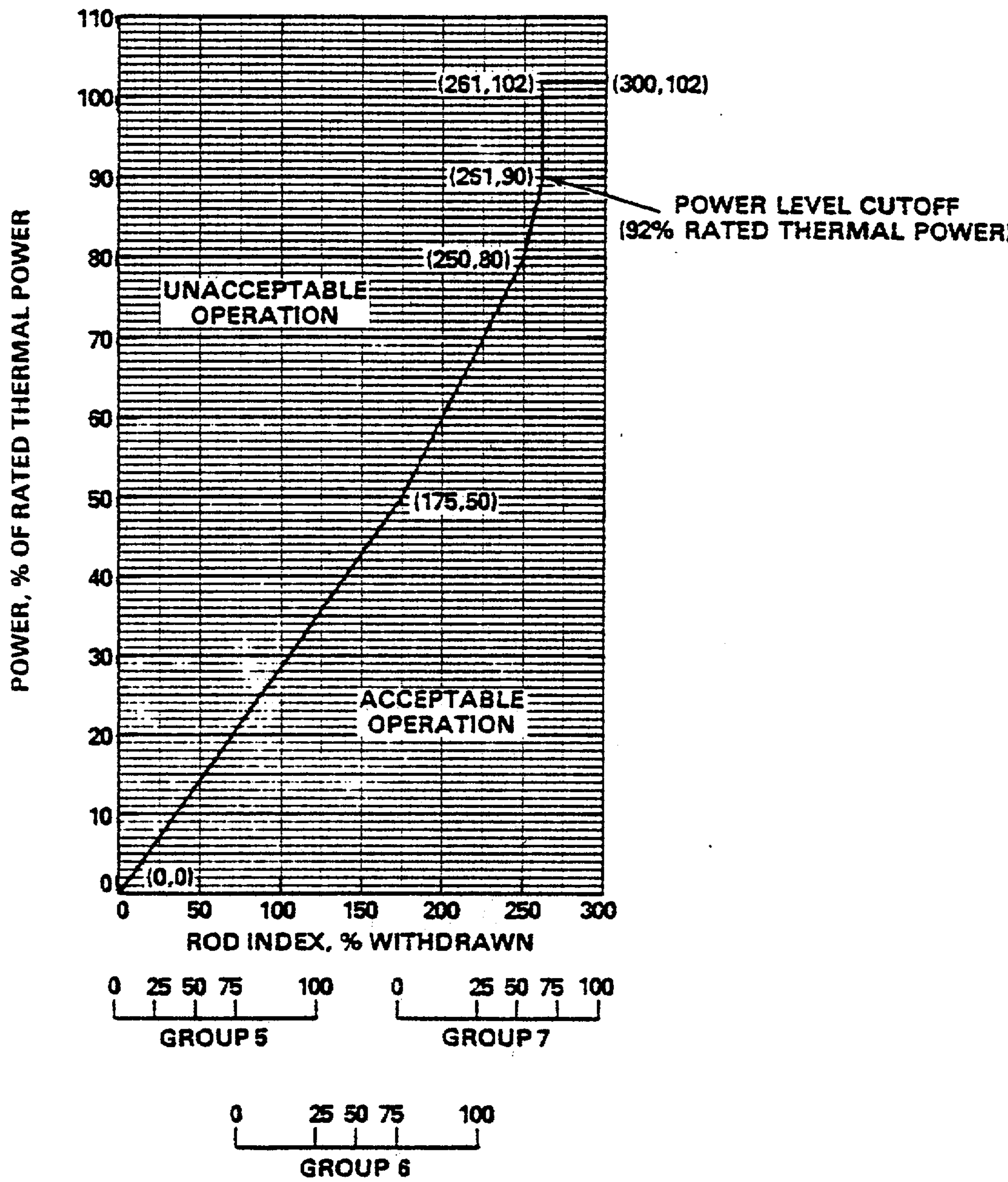


FIGURE 3.1-2

REGULATING ROD GROUP INSERTION LIMITS FOR 4 PUMP OPERATION AFTER 250 ± 10 EFPD

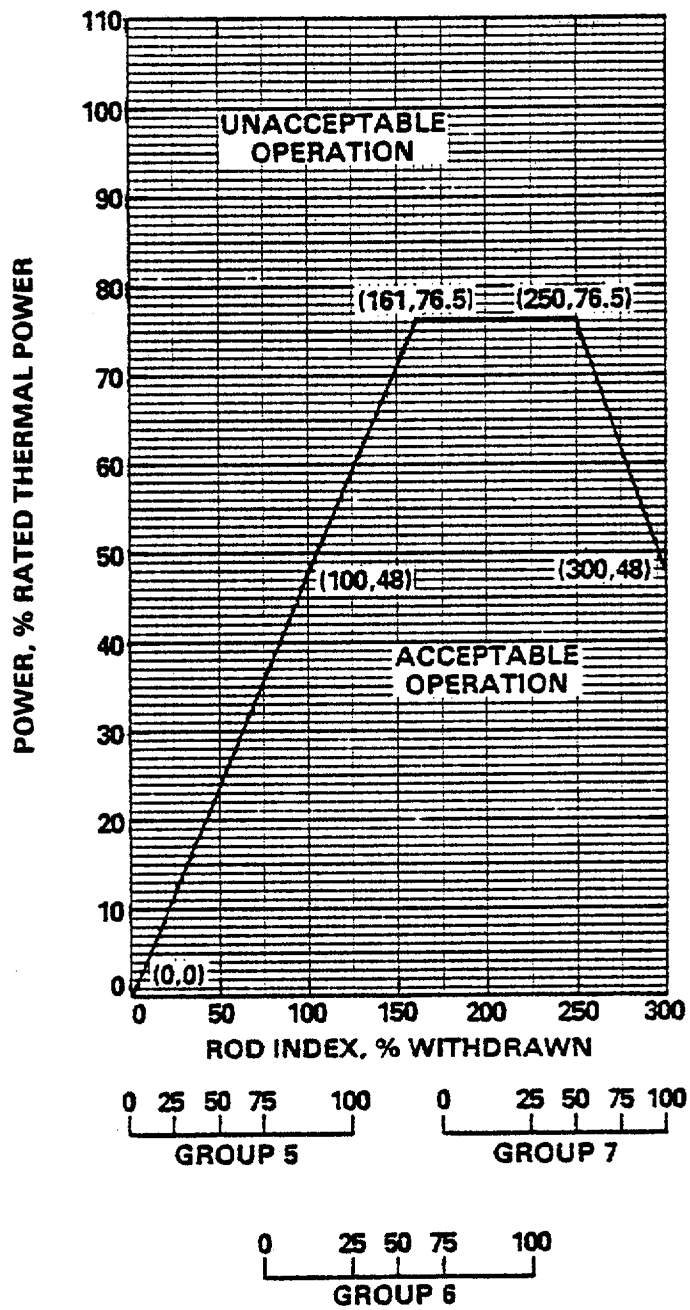


FIGURE 3.1-3

REGULATING ROD GROUP INSERTION LIMITS FOR 3 PUMP OPERATION FROM 0 EFPD TO 250 ± 10 EFPD

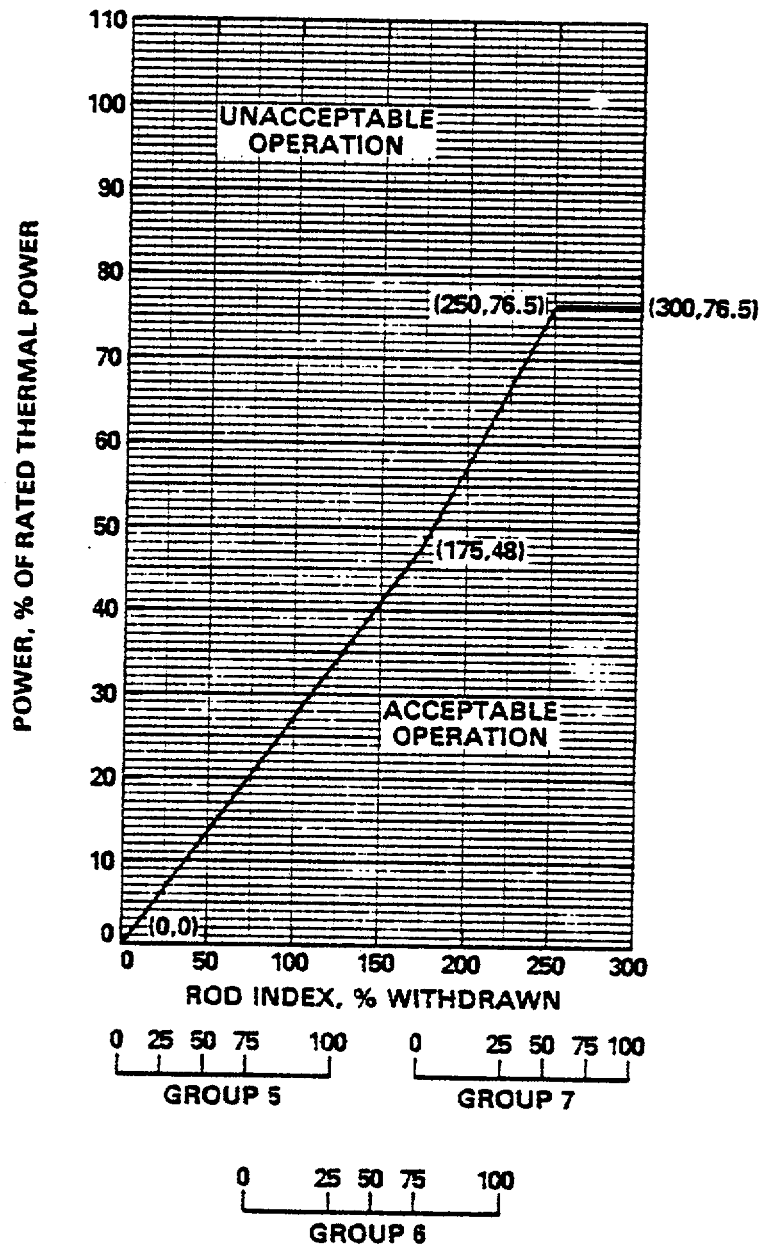


FIGURE 3.1-4

REGULATING ROD GROUP INSERTION LIMITS FOR
3 PUMP OPERATION AFTER 250 ± 10 EFPD

REACTIVITY CONTROL SYSTEMS

AXIAL POWER SHAPING ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

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

APPLICABILITY: MODES 1 and 2*.

ACTION:

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

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

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

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

SURVEILLANCE REQUIREMENTS

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

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

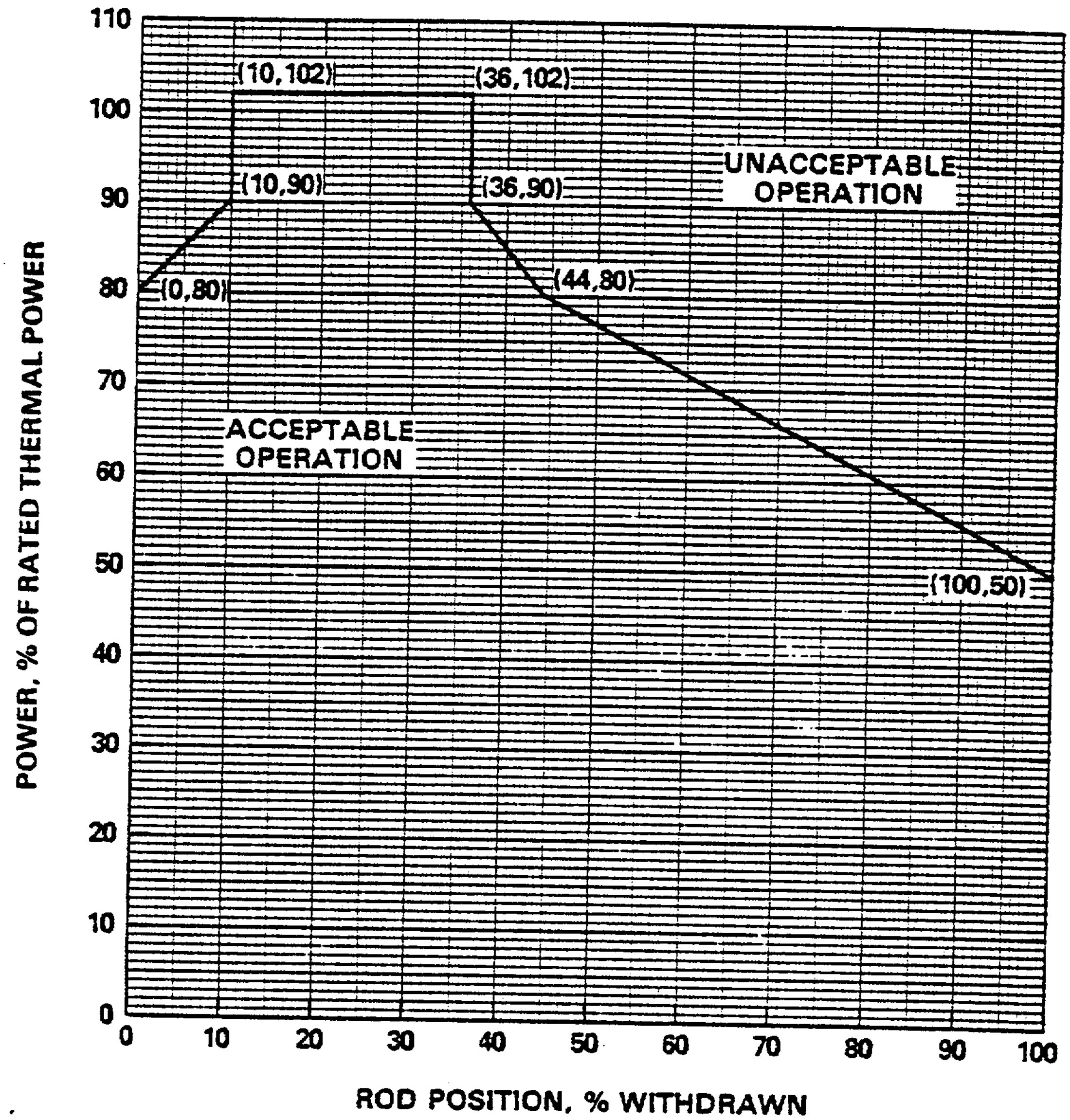


FIGURE 3.1-9

AXIAL POWER SHAPING ROD GROUP INSERTION LIMITS FROM 0 EFPD TO 250 ± 10 EFPD

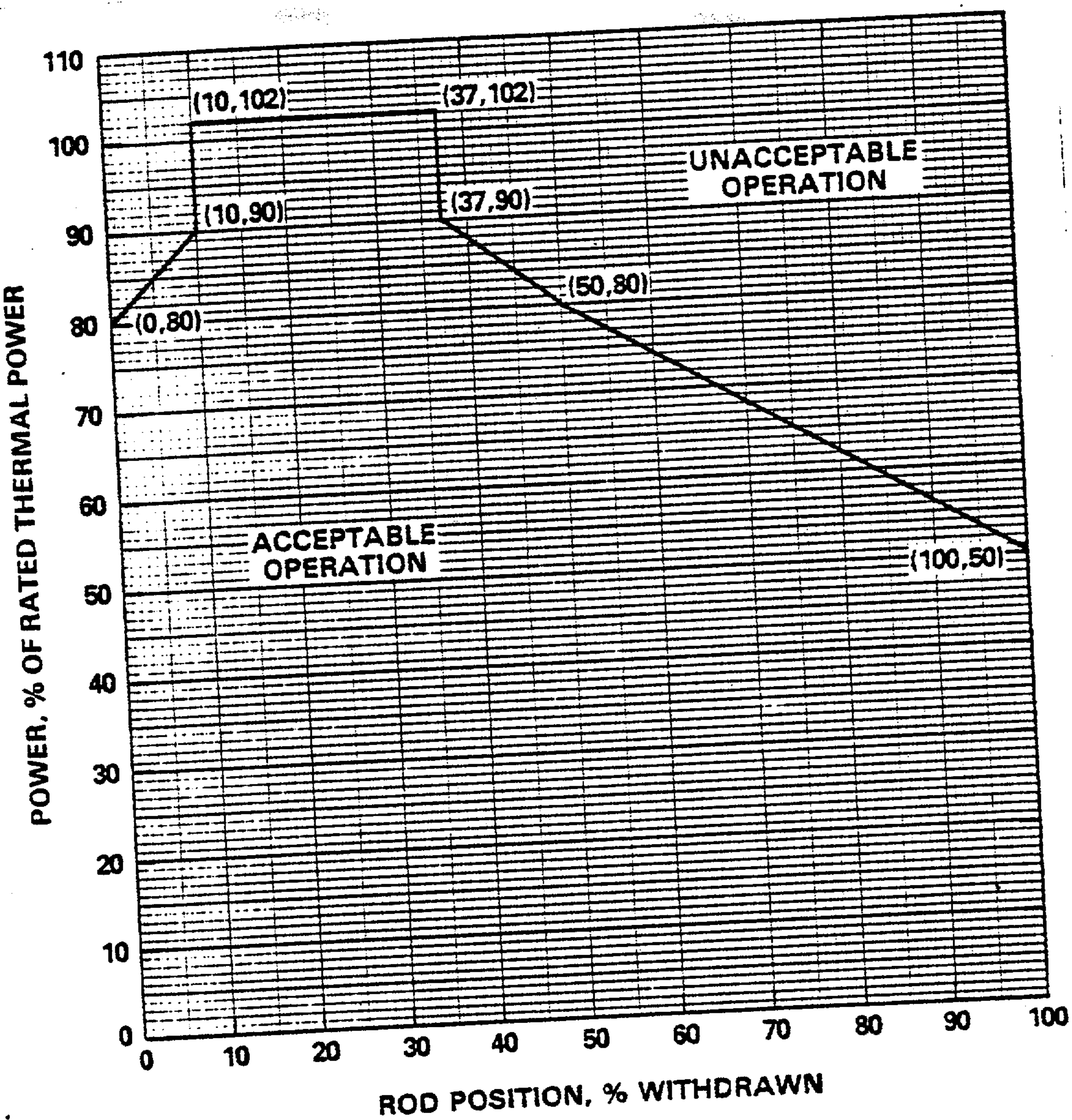


FIGURE 3.1-10

AXIAL POWER SHAPING ROD GROUP INSERTION LIMITS AFTER 250 ± 10 EFPD

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.

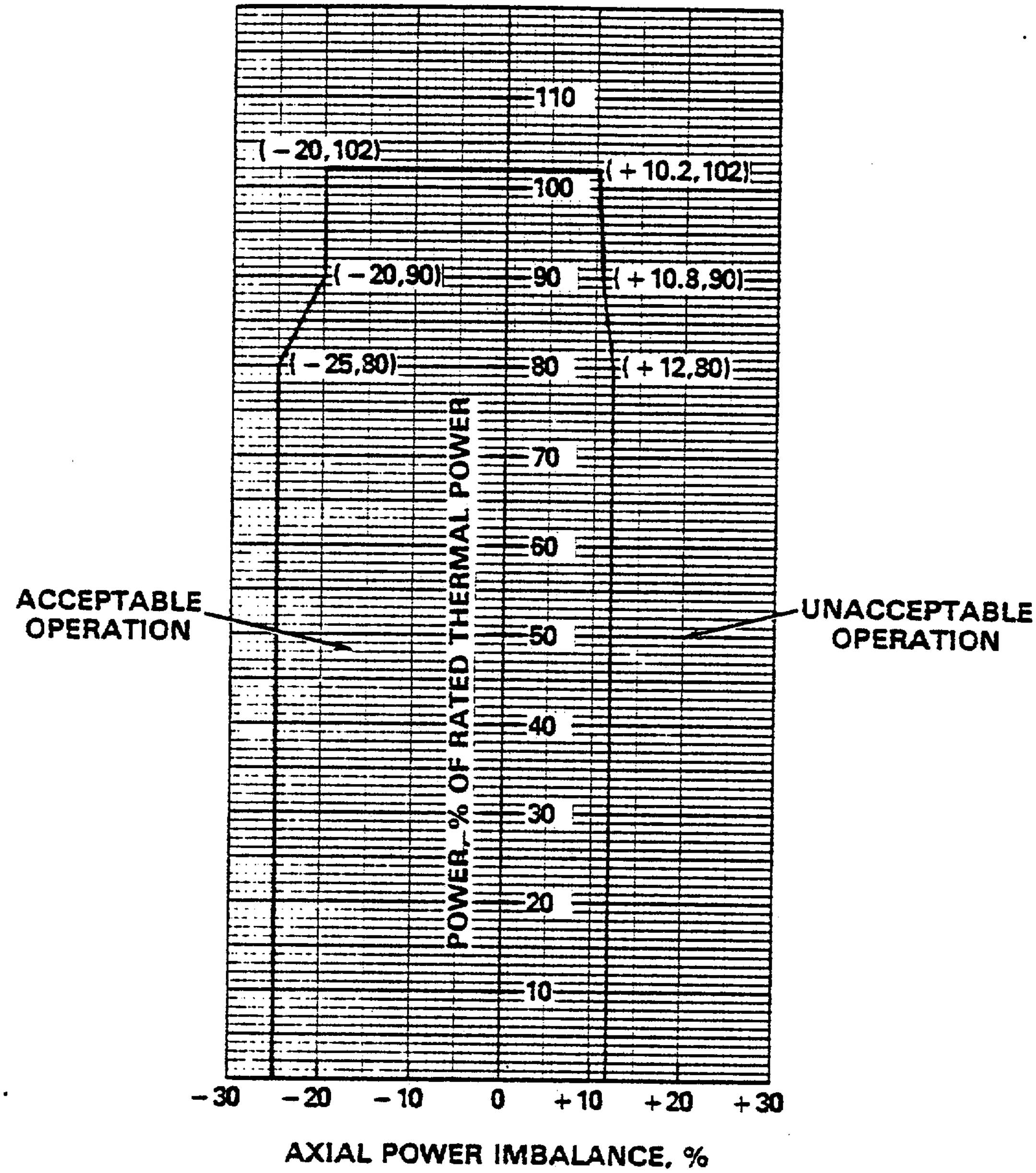


FIGURE 3.2-1

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION FROM 0 EFPD TO 250 ± 10 EFPD

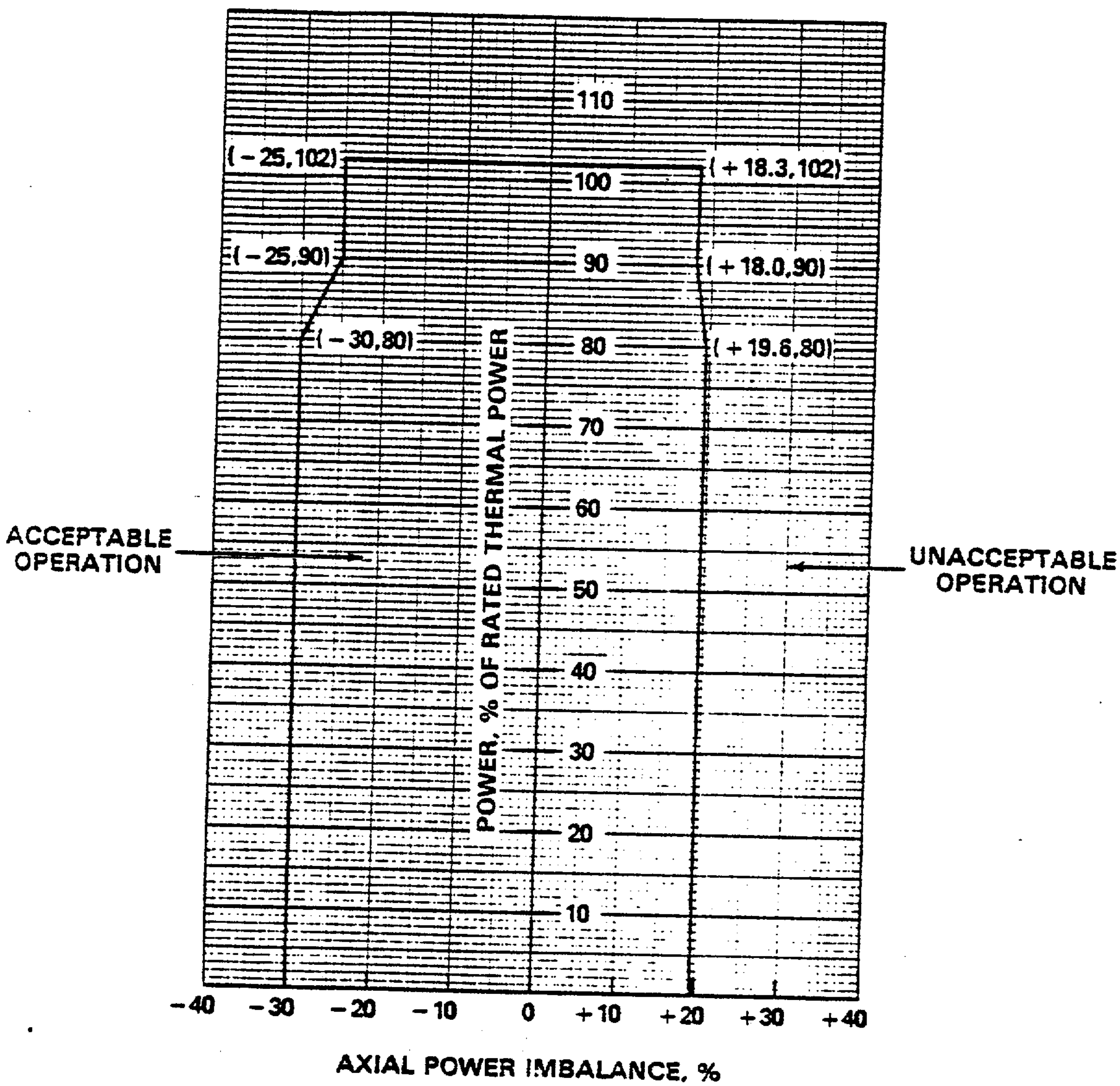


FIGURE 3.2-2

AXIAL POWER IMBALANCE ENVELOPE FOR OPERATION AFTER 250 ± 10 EFPD

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:

TABLE 3.2-2

QUADRANT POWER TILT LIMITS

	STEADY STATE LIMIT	TRANSIENT LIMIT	MAXIMUM LIMIT
QUADRANT POWER TILT as Measured by:			
Symmetrical Incore Detector System	3.31	8.81	20.0
Power Range Channels	1.96	6.96	20.0
Minimum Incore Detector System	1.90	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 3.2-1

DNB MARGIN

Parameter	LIMITS Four Reactor Coolant Pumps Operating	 Three Reactor Coolant Pumps Operating
Reactor Coolant Hot Leg Temperature, T_H °F	$\leq$ 604.6	$\leq$ 604.6(1)
Reactor Coolant Pressure, psig(2)	$\geq$ 2061.6	$\geq$ 2057.2(1)
Reactor Coolant Flow Rate, lb/hr	$\geq 139.7 \times 10^6$	$\geq 104.4 \times 10^6$

(1) Applicable to the loop with 2 Reactor Coolant Pumps Operating.

(2) Limit not applicable during either a THERMAL POWER ramp increase in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step increase of greater than 10% of RATED THERMAL POWER.

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 77.98% of RATED THERMAL POWER and within 4 hours the setpoints for the following trips have been reduced to the values specified in Specification 2.2.1 for operation with three reactor coolant pumps operating:

    1. Nuclear Overpower

MODES 3, 4 and 5:

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

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

SURVEILLANCE REQUIREMENTS

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

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

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

*See Special Test Exception 3.10.3.

INTENTIONALLY BLANK

SPECIAL TEST EXCEPTION

NO FLOW TEST

LIMITING CONDITION FOR OPERATION

3.10.3 The limitations of Specification 3.4.1 may be suspended during the performance of startup and PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Nuclear Overpower channels are set $\leq$ 25% of RATED THERMAL POWER.

APPLICABILITY: During startup and PHYSICS TESTS.

ACTION:

With the THERMAL POWER greater than 5% of RATED THERMAL POWER, immediately open the control rod drive trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.3.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour during startup and PHYSICS TESTS.

4.10.3.2 Each Nuclear Overpower Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating startup or PHYSICS TESTS.

SPECIAL TEST EXCEPTION

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.4 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided:

a. Reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s), and

b. All axial power shaping rods are withdrawn to at least 35% (indicated position) and OPERABLE.

APPLICABILITY: MODE 2.

ACTION:

a. With any safety or regulating control rod not fully inserted and with less than the above reactivity equivalent available for trip insertion or the axial power shaping rods not within their withdrawal limits, immediately initiate and continue boration at ≥ 10 gpm of 11,600 ppm boric acid solution or its equivalent, until the SHUTDOWN MARGIN required by Specification 3.1.1.1.1 is restored.

b. With all safety or regulating control rods fully inserted and the reactor subcritical by less than the above reactivity equivalent, immediately initiate and continue boration at ≥ 10 gpm of 11,600 ppm boric acid solution or its equivalent, until the SHUTDOWN MARGIN required by Specification 3.1.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The position of each safety, regulating, and axial power shaping rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.4.2 Each safety or regulating control rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.1.

4.10.4.3 The axial power shaping rods shall be demonstrated OPERABLE by moving each axial power shaping rod ≥ 6.5% (indicated position) within 4 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.1.

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.

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 1) the moderator temperature coefficient is within its analyzed temperature range, 2) the protective instrumentation is within its normal operating range, 3) the pressurizer is capable of being in an OPERABLE status with a steam bubble, and 4) the reactor pressure vessel is above its minimum RT_{NDT} temperature.

3/4.1.2 BORATION SYSTEMS

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

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

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 3.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires either 6615 gallons of 11,600 ppm boric acid solution from the boric acid storage tanks or 45,421 gallons of 2270 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.

3/4.1.2 BORATION SYSTEMS (Continued)

The boron capability in Modes 4 and 5 is based on a potential moderator dilution accident and is sufficient to provide a SHUTDOWN MARGIN of 3.0% Δk/k after xenon decay and a 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 1608 gallons of 2270 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 containment after a design basis accident. The pH band minimizes the evolution of iodine and minimizes the effect of chloride and caustic stress corrosion cracking on mechanical systems and components.

The 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 effects 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 THERMAL POWER based on xenon reactivity is necessary to ensure that power peaking limits are not exceeded even with specified rod insertion limits 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. 32 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 letters dated March 21, 1980, April 14 and 30, 1980, June 6 and 13, 1980, and July 22 and 31, 1980, Florida Power Corporation (FPC or licensee) requested amendment of Appendix A to the Crystal River-3 (CR-3) Operating License to permit power operation during Cycle 3.

Reference 5-2 includes a Babcock & Wilcox Company (B&W) report BAW-1607, Rev. 1, dated April 1980 to support the CR-3 Cycle 3 operation at an upgraded power level of 2544 MWt as opposed to Cycle 2 power level of 2452 MWt. The report describes the fuel system design, nuclear design, thermal-hydraulic design, accident analysis, and startup test program. The report supports Cycle 3 operation for 335 effective full power days (EFPD) at the upgraded power level.

Our letter of April 4, 1980 stated we couldn't consider a power increase to 2544 MWt until we resolve our concerns about sensitivity to secondary side transients and other post TMI-2 concerns. By a letter dated April 30, 1980 (Reference 4-4), FPC informed the NRC of their decision to operate at the currently licensed power level of 2452 MWt when Cycle 3 is started. Also, in Reference 4-4, FPC submitted modified technical specifications that reflect Cycle 3 operation without the power upgrade. The reason given by FPC for not going to the higher power level at this time is the unavailability of the reactor coolant pump power monitors (RCPPMs). The RCPPMs are required for the upgraded power operation so that in a postulated loss of coolant flow (LOCF) event, the RCPPMs will trip the reactor sooner than the existing flux/flow comparator. The faster RCPPM response (0.62 sec as compared to 1.40 sec for the flux/flow comparator) decreases the time during which the reactor is operating at 2544 MWt with degrading flow conditions (LOCF), so that the departure from nucleate boiling ratio (DNBR) criterion is not violated.

Even though no power increase will be implemented when the plant resumes operation in Cycle 3, this safety review and accident analyses evaluation are based on the higher power level of 2544 MWt. Sections 2.0, 3.0, 4.0, and 5.0 below evaluate the Fuel System Design, Nuclear Design and Startup Test Program, Thermal-Hydraulic, and Accidents and Transients respectively. The technical specifications for Cycle 3 operation at the lower power level of 2452 MWt are evaluated at the end of sections 3.0 and 4.0 below. References are listed at the end of each section.

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2. Evaluation of Fuel System Design

2.1 Fuel Assembly Mechanical Design

The fresh Babcock and Wilcox Mark B-4 fuel assemblies loaded as Batch 5 at the end of Cycle 2 (EOC 2) are mechanically interchangeable with Batches 2 and 3 (Mark B-3) and Batch 4 (Mark B-4) fuel assemblies previously loaded at Crystal River Unit 3. Fifty-two Batch 2 assemblies and four Batch 4 assemblies have been discharged and fifty-six Batch 5 assemblies will be loaded for Cycle 3. This reload scheme is a revision (2-1) to that originally proposed (2-2) by the licensee. The change allows the replacement of one Batch 4 assembly with a broken hold-down spring and three additional, symmetric, Batch 4 assemblies. Our evaluation of the broken hold-down spring is further discussed in Section 2.4.2.

The Mark B-4 fuel assembly has been previously approved (2-3) by the NRC staff and is utilized in other B&W nuclear steam supply systems. The new assemblies have modified end fittings, mainly to reduce the coolant flow pressure drop. The Mark B-4 assemblies also incorporate some modification to the spacer grid corner cells to reduce wear during fuel handling. Two assemblies will contain primary neutron sources and two assemblies will contain regenerative neutron sources in Cycle 3. The justification (2-4) for the design of the retainer is applicable to the neutron sources used in Cycle 3.

2.2 Fuel Rod Design

Although Crystal River 3 Batch 4 and Batch 5 utilize the same Mark B-4 fuel, the Batch 5 assemblies incorporate a slightly higher initial fuel density. The change, from 94 to 95 percent of theoretical density, is a consequence of using a more stable (densification resistant) fuel material. This change results in a shorter initial, but almost identical densified, active fuel length.

The fuel pellet end configuration has also changed to a truncated cone dish for Batch 5 as opposed to a spherical dish for the previous four batches. This minor change facilitates manufacturing and does not significantly alter the performance characteristics of the fuel.

2.2.1 Cladding Collapse

Due to the cumulative nature of cladding deformation, creep collapse analyses were performed for the previous two cycles as well as the proposed third cycle of operation. Batches 2 and 3 are more limiting than Batches 4 and 5 due to their previous incore exposure time. That analysis was performed for the most limiting fuel assembly power history using the CROV computer code and procedures described in the topical report BAW-10084PA, Rev. 2 (2-5). The analysis conservatively determined a creep collapse time of 25,000 effective full power hours (EFPH) of operation. Since the collapse time is greater than the estimated residence time for the most limiting assembly at EOC3 (22,800 EFPH), we conclude that cladding creep collapse has been adequately considered.

2.2.2 Cladding Stress

The licensee stated (2-2) that the Batch 2 and 3 reinserted fuel assemblies are the limiting batches from a cladding stress point of view because of their lower density and longer previous exposure time.

We have examined the mechanical analysis section of the CR-3 Fuel Densification Report and find the cladding stress analyses were performed for both beginning-of-life and end-of-life (EOC-3) conditions for first cycle fuel. The results, shown in Table 3.5-1 of the report, compare cladding circumferential stress levels with the yield and ultimate strength of Zircaloy under a variety of conditions. The cladding stress levels are strongly dependent on the pressure differential across the cladding wall and are limiting (maximum) for beginning-of-life when the rod internal pressure is minimum. This is contrary to the exposure dependence cited in Reference 2-6. In addition, we find no evidence that the lower density fuels are limiting in the analyses.

We agree that the pressure differential across the cladding wall is a major contributor to the cladding stress level. The external system pressure remains relatively constant (2200 psia) during normal operation. The differential across the cladding wall is the greatest, therefore, when the rod internal pressure is much less, or much greater, than the coolant pressure. As discussed in Section 2.2.4, the rod internal pressure does not exceed system pressure during normal operation. Therefore, limiting cladding stress conditions based on rod internal gas pressure exist at beginning-of-life. The licensee has informed us that Batches 4 and 5 fuel have a higher initial fill gas pressure than Batches 2 and 3. As a result, the analyses presented in the CR-3 densification report may be applied to Cycle 3 operation.

We also note, however, that fuel swelling, cladding creep, and fuel-cladding mechanical interaction may also contribute to the effects of internal gas pressure on cladding stress levels. In general, these effects are localized and the licensee's design bases (CR-3 FSAR 3.1.2.4.2 (2-7)) state that such "stresses relieved by small material deformation are permitted to exceed the yield strength." We do not believe that the design criterion for cladding stress will limit the operational flexibility of CR-3. Therefore, we conclude that cladding stress limits, will not be exceeded during normal operation of Cycle 3 fuel at CR-3.

2.2.3 Cladding Strain

The fuel design criteria (CR-3 FSAR Section 3.1.2.4.2 (2-7)) specify a 1% limit on cladding plastic strain due to diameter increases resulting from fuel swelling, thermal ratcheting, creep and internal gas pressure. Strain limits were established on the basis of low-cycle fatigue techniques, not to exceed 90% of material fatigue life. The design evaluation, discussed in Section 3.2.4.2.1 of the CR-3 FSAR (2-7) and Section 3.5.2 of the CR-3 Fuel Densification Report (2-6), was performed for design pellet burnup and heat generation rate as well as limiting dimensional tolerances. These conditions are considerably beyond those expected for Cycle 3 at Crystal River Unit 3. The results show circumferential plastic strain is less than 1% at design EOL burnup, and cumulative fatigue damage after three cycles of operation is less than 90% of material fatigue life. We conclude that the cladding strain and fatigue limits have been adequately considered for Cycle 3 operation.

2.2.4 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (2-8) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Not all of these have been addressed in the licensee's reload application or previous reports. Among those which may affect the operation of the fuel rod is the internal pressure limit. Our current criterion (SRP 4.2, Section II.A.1(f)) states that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified. Meeting this criterion is also a condition of acceptance as discussed previously in Section 2.2.2 (cladding stress).

Although the CR-3 FSAR states that "at end-of-life, fission gas pressure does not exceed system pressure" (2-7), it also describes the use of an internal gas pressure of 3,300 psi to determine fuel cladding internal design conditions. It is not clear whether the limit of rod internal pressure on system pressure is a design criterion or simply an analytical result. The analysis is not described in the reload submittal. Furthermore, we believe (2-9) that some of the analytical methods utilized by Babcock and Wilcox may be deficient at high burnups.

In response to a question of this criterion, Florida Power has stated (2-10) that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 3. This analysis is based on the use of the B&W TAFY code (2-11) rather than a newer B&W code called TACO (2-12). Although both of these codes are currently approved for use in safety analyses, we believe 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 (2-13) 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/TU. Although we have not examined the comparison, we note that the analyses exceed the expected exposure in CR-3 Cycle 3 by a large margin. We conclude that the rod internal pressure limits have been adequately considered.

2.3 Fuel Thermal Design

There are no major changes between the new Batch 5 fuel and previous batches reinserted in the Cycle 3 core. The increase in initial fuel density (95% T.D.) results in a slightly higher linear heat rating for the fuel based on centerline melt. The rating was established with the TAFY code (2-11). We have performed an independent check of the Batch 5 fuel design parameters and agree that fuel melting will not occur for the linear heat ratings given in Table 4.2 of Ref. 2-2.

The average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the LOCA analysis (Section 7.15 of the Reload submittal) are also calculated with the TAFY code (2-11). Babcock and Wilcox has stated (2-2) that the fuel temperature and pin pressure data used in the generic LOCA analysis (2-14) are conservative compared to those calculated for Cycle 3 at Crystal River 3.

As previously mentioned in Section 2.2.4 of this evaluation, B&W currently has two fuel performance codes, TAFY (2-11) and TACO (2-12), which could be used to calculate the LOCA initial conditions. The older code TAFY has been used for the Cycle 3 LOCA analysis. Recent information (2-15) indicates that the TAFY code predictions do not produce higher peak cladding temperatures than TACO for all Cycle 3 conditions as suggested in Ref. 2-13. 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. Babcock and Wilcox has proposed (Attachment 3 of Ref. 2-13) a method of resolving this issue which has not yet been accepted by the staff. While we have not yet completed the review, we believe the Cycle 3 LOCA initial conditions are acceptable as submitted.

2.4 Operating Experience

Babcock & Wilcox 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 is given on page 4-3 of Ref. 2-2.

2.4.1 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 those facilities designed by Westinghouse. In a letter dated June 13, 1978, we requested information from Babcock and Wilcox 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 was documented in our letter to B&W dated August 22, 1979.

Because significant guide tube wear could impede the control rod scram capability, and also affect the required coolable geometry of the reactor core, we consider this wear phenomenon a potential safety concern. Therefore, we requested (2-16) additional information from the licensee on the wear characteristics of the control rods on the guide tubes at CR-3. The response to this request has not yet been received. The licensee has stated (2-10) that a generic response to this request has been prepared by Babcock and Wilcox. The report, B&W Control Rod Guide Tube Wear Generic Report (BAW-1623), has been concurred with by the licensee but has not been received by the NRC.

We have, however, received preliminary information on post-irradiation examinations of identical guide tubes for wear in Rancho Seco spent fuel (2-17). The results of these measurements indicate that through-wall wear or excessive wall degradation will not likely occur during anticipated fuel residence time under control rod assemblies. On the basis of this preliminary information and the imminent documentation of a complete generic evaluation, we conclude that guide tube wear has been adequately addressed for Crystal River 3 during Cycle 3.

2.4.2 Hold-Down Spring Damage

Davis-Besse Unit 1, another B&W designed reactor, reported fuel assembly hold-down spring damage in late May of this year. Due to the similarity of the reactor and fuel assemblies used at Crystal River Unit 3, all in-core and discharged fuel assemblies were examined for hold-down spring damage. A broken hold-down spring was discovered in assembly NJ018E, a Batch 4 assembly that had been in core location N-14 during Cycle 2. This assembly and three symmetric assemblies were replaced with Batch 5 fuel. The resulting changes to the Crystal River Unit 3 Cycle 3 Reload Report (2-2) are discussed in Reference 2-1.

2.5 Rod Bow

The licensee has stated that a rod bow penalty has been calculated according to the procedure approved in reference 2-17. The burnup used is the maximum fuel assembly burnup of the batch that contains the limiting (maximum radial x local peak) fuel assembly. For Cycle 3, this burnup is 31,358 MWD/MTU in a Batch 3 assembly. The resultant net rod bow penalty after inclusion of the 1% flow area reduction factor credit is 2.8% reduction in DNBR. However, this rod bow penalty is offset by the 10.2% DNBR margin included in trip setpoints and operating limits.

2.6 Densification Power Spike

The densification power spike was eliminated from DNBR evaluations based on the NRC approval of this change in reference 2-19.

2.7 Cladding Strain and Flow Blockage

The licensee has responded (2-20) to our request for information concerning the new fuel cladding strain and fuel assembly flow blockage models described in NUREG-0630.

Florida Power Corporation has reviewed all of the subject information supplied by Babcock & Wilcox and is in agreement with the results that calculated peak fuel cladding temperature will remain unchanged or lowered with the use of the new NRC ramp-rate-dependent correlations, and that compliance with 10 CFR 50.46 is assured for Crystal River Unit 3.

2.8 References

2-1 P. Y. Baynard (Florida Power) letter to Director, Office of Nuclear Reactor Regulation (NRC), dated June 6, 1980.

2-2 Crystal River Unit 3 - Cycle 3 Reload Report, Babcock and Wilcox Company Report BAW-1607, February 1980.

2-3 R. W. Reid (NRC) letter to W. P. Stewart (Florida Power), dated July 3, 1979.

2-4 BPRA Retainer Design Report, Babcock and Wilcox Company Report BAW-1496, May 1978.

2-5 Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, Babcock and Wilcox Company Report BAW-10084P-A, October 1978.

2-6 Crystal River Unit 3 Fuel Densification Report, Babcock and Wilcox Company Report BAW-1397, August 1973.

2-7 Crystal River Unit 3 Nuclear Generating Plant Final Safety Analysis Report, Docket No. 50-302, Florida Power Corporation, March 1977.

2-8 Standard Review Plan, Section 4.2 (Rev. 1), "Fuel System Design," U.S. Nuclear Regulatory Commission Report NUREG-75/087.

2-9 J. Stolz (NRC) letter to Florida Power Corporation, dated February 11, 1977.

2-10 R. M. Bright (Florida Power) letter to Office Director of Nuclear Reactor Regulation (NRC), dated June 13, 1980.

2-11 C. D. Morgan and H. S. Kao, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis," Babcock and Wilcox Company Report BAW-10044, May 1972.

2-12 "TACO-Fuel Pin Performance Analysis, Babcock and Wilcox Company Report BAW-10087P-A, Rev. 2, August 1977.

2-13 J. H. Taylor (B&W) letter to P. S. Check (NRC), dated July 18, 1978.

2-14 R. C. Jones, J. R. Biller, and B. M. Dunn, "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," Babcock and Wilcox Company Report BAW-10103A, Rev. 3, July 1977.

2-15 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.

2-16 R. W. Reid (NRC) letter to W. P. Stewart (Florida Power) dated November 23, 1979.

2-17 J. J. Mattimoe (Sacramento Municipal Utility District) letter to R. W. Reid (NRC) on "Guide Tube Wear Measurements--Preliminary Results," February 15, 1980.

2-18 L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) on "Evaluation of Interim Procedure for Calculating DNBR Reductions due to Rod Bow," dated October 18, 1979.

2-19 S. A. Varga (NRC) letter to J. H. Taylor (B&W) on "Update of BAW-10055, Fuel Densification Report," December 5, 1977.

2-20 J. A. Hancock (Florida Power) letter to D. G. Eisenhut (NRC) dated December 20, 1979.

3.0 Evaluation of Nuclear Design and Startup Test Program

3.1 General

A core loading diagram for Cycle 3 is presented in the reload report (BAW 1607, Revision 1) along with enrichment and burnup distributions. The nuclear parameters for Cycle 3 are compared to those for Cycle 2 including reactivity coefficients, boron worths and rod group worths. An analysis of the shutdown margin capability and a radial power map at BOC are also given.

The core physics calculations are performed with PDQ07 code (Reference 3-1) which has been reviewed and approved by the staff. This code has been used for analysis of the previous cycles of CR-3. The results of the analysis show small differences between Cycle 3 and Cycle 2 values, occasioned by the difference in cycle lengths (335 EFPD for Cycle 3 vs. 275 EFPD for Cycle 2) and by the fact that the core is not yet in its equilibrium configuration. The analysis of shutdown margin shows that 1.84% Δ k/k exists at end of cycle compared to the required 1.0% Δ k/k for hot shutdown. The calculated radial power distribution at BOC shows adequate margin to limits.

Based on the fact that approved methods have been used to obtain the core characteristics, that margin exists to limiting values of the parameters, and that startup testing will be used to obtain measured values of important parameters, we find the analysis of core parameters to be acceptable.

3.2 Evaluation of Fuel Loading Error and Rod Misoperation Transients

The accident and transient analyses presented in the FSAR have been examined to determine the effect of increasing core power level to 2544 MWt. The results of this examination were evaluated as part of the review of the Cycle 2 reload submittal. The results of that evaluation are presented in Table 1 of the Safety Evaluation (Reference 3-2) for that submittal. The conclusions reached in that table still apply to the rod withdrawal error, rod misoperation, fuel loading error, and rod ejection events.

3.3 Startup Test Program

The physics startup test program as submitted by the licensee in BAW 1607 has been revised (Reference 3-4). The final program is identical to that used for Cycle 2 with the exception of a revised review criterion applied to the power distribution measurements.

The program consisted of zero-power test and power escalation test. The zero-power test consisted of (a) critical boron concentration, (b) temperature reactivity coefficient, (c) control rod group reactivity worth, (d) ejected control rod reactivity worth measurements, and (e) a symmetry test involving swapping of symmetrical rods.

The power escalation tests consisted of (a) core power distribution verification at 40%, 75% and 100% full power, (b) incore vs. excore detector imbalance correlation verification, (c) temperature reactivity coefficient, and (d) power Doppler reactivity coefficient measurements.

The staff has reviewed the complete physics startup test program including review and acceptance criteria and remedial actions and finds this program acceptable.

3.4 Evaluation of Power Distribution and Reactivity Technical Specification Changes

We have reviewed Figures 2.1-2, 2.1-1, 3.1-1, 3.1-2, 3.1-3, 3.1-4, 3.1-9, 3.1-10, 3.2-1 and 3.2-2 and Tables 2.2-1 and 3.2-2 of the proposed Technical Specifications (Reference 3-3). The same procedures and techniques were employed to derive these curves and tables as have been used for previous cycles. The changes from the previous cycle curves are not large and are consistent with the changes in core parameters. On these bases we find the above cited changes to the Technical Specifications to be acceptable.

3.5 References

3.1 PDQ07 Users Manual, BAW-10117 PA, January, 1977.
3.2 Letter, R. W. Reid, NRC to W.P. Stewart, FPC, July 3, 1979 with attachments.
3.3 Letter, R. M. Bright, FPC, to Director, NRR, NRC, April 30, 1980.
3.4 Letter, R. M. Bright, FPC, to Director, NRR, NRC, June 13, 1980.

4.0 Evaluation of Thermal Hydraulic Design

4.1 DNBR Evaluations

A comparison between the thermal hydraulic design conditions for Cycles 1, 2 and 3 (Reference 4-1) is listed in Table 4.1. The design power level for Crystal River 3 Cycle 3 reload is 2544 MWt even though it will actually operate at the licensed core power level of 2452 MWt. However, the thermal hydraulic design calculations in support of Cycle 3 operation assumed a power level of 2568 MWt (same as for Cycle 2) for consistency with other B&W plants. A summary of our evaluation follows:

(a) Critical Heat Flux - The B&W-2 critical heat flux (CHF) correlation in conjunction with the TEMP thermal hydraulic code (Reference 4-2) was used for DNBR evaluation instead of the W-3 correlation used for Cycle 1. The B&W-2 correlation has been approved by the staff and is currently used to license all operating B&W plants with Mark-B fuel assembly cores including the Crystal River 3, Cycle 2. (Reference 4-3).

(b) Reactor Coolant Flow - The assumed system flow for Cycle 3 analyses is 106.5% of the design flow (88,000 gpm/pump) and is the same as the low flow limit included in the Technical Specifications and analyses for Cycle 2. The flow rate from measurements at Crystal River 3 indicate a system flow capability of 109.5% of design flow rate, including measurement uncertainty.

(c) Rod Bow - As discussed in Section 2.5, a net rod bow penalty of 2.8% reduction in DNBR has been calculated by approved methods. This is acceptable for Cycle 3 operation since a 10.2% DNBR margin, exclusive of the penalty is available.

(d) Peaking Factor - The licensee has stated that a reference design radial x local power peaking factor ($F_{\Delta H}$) of 1.71 was used for Cycle 2 and 3 evaluations. The Cycle 1 $F_{\Delta H}$ of 1.78 was reduced to 1.71 in conjunction with orifice rod assembly and burnable poison rod assembly removal.

4.2 Pressure-Temperature Limit Analysis

The licensee presented pressure-temperature limit curves for four and three pump operation. The most limiting of these curves (four-pump) provides the basis for the reactor protection system variable low-pressure trip function. The curves are based on a minimum DNBR of 1.433, which provides 10.2% margin to the CHF correlation limit and allows flexibility for future cycle designs.

4.3 Loss-of-Coolant-Flow-Transients

The flux/flow trip is designed to protect the plant during pump coastdowns from four-pump operation or to act as a high flux trip during partial-pump operation. Redundant pump monitors will be installed for each Crystal River 3 pump prior to operation at 2544 MWt in order to trip the reactor immediately upon the loss of power to two or more pumps. The flux/flow trip setpoint will then serve only to protect the plant during a one-pump coastdown from four-pump operation. The licensee stated that the margin for flux/flow setpoint was determined with a transient analysis initiated from 108% instead of 102% of 2544 MWt. This margin allows for uncertainties in power measurements and heat balance error. While the analyses are acceptable for operation at 2544 MWt, the indicated power measurement uncertainties (6%) imposed on a 102% real power level infers an operating mode that would be unacceptable. The licensee will not be permitted to operate the plant with the sustained indicated power level above the license limit (100%). Operation at 100% is acceptable if the power measurement uncertainties are not greater than ± 2%, giving a maximum real power level of 102%.

For the analysis, actual measured one-pump coastdown data were used and maximum additive trip delays were used between the time trip conditions were reached and actual control rod motion started. Once a flux/flow trip limit was found to be adequate by thermal-hydraulic analysis, error adjustments were made to account for flow measurement noise and instrument error before the actual trip setpoint was determined. The staff finds these analyses methods to be acceptable. The recommended Cycle 3 thermal-hydraulic flux/flow trip limit of 1.10 (actual in-plant setpoint of 1.07) resulted in a transient minimum DNBR of 1.75 (B&W-2) during the pump coastdown. This represents >34% DNBR margin to the correlation limit of 1.30, and is therefore acceptable.

The four-pump coastdown and locked rotor transients were analyzed at a power level of 102% of 2568 MWt. The results are discussed in Section 5, "Evaluation of Accidents and Transients".

Table 4.1 Cycle 1, 2 and 3 Thermal-Hydraulic Design Conditions

	Cycle 1 <268.8 EFPD	Cycle 1 >268.8 EFPD	Cycles 2 & 3 2544 MWt
Design power level, MWt	2452	2452	2568
System pressure, psia	2200	2200	2200
Reactor coolant design flow, gpm % design	352,000	352,000	352,000
Reactor Coolant Flow, % design	105	105	106.5
Ref design radial x local power peaking factor, $F_{\Delta H}$	1.78	1.71	1.71
Ref design axial flux shape	1.5 cosine	1.5 cosine	1.5 cosine
Hot channel factors			
Enthalpy rise	1.011	1.011	1.011
Heat flux	1.014	1.014	1.014
Flow area	0.98	0.98	0.98
Densified active length, in.	141.12	140.2[b]	140.2[b]
Avg heat flux at 100% power, Btu/h-ft^2	167×10^3	168×10^3	176×10^3
Max heat flux at 100% power, Btu/h-ft^2	446×10^3[a]	431×10^3	452×10^3
CHF correlation	W-3	B&W-2	B&W-2
Minimum DNBR (% power)	1.61 (114) 1.92 (102)	2.14 (112) 2.27 (108) 2.49 (102)	1.98 (112) 2.12 (108) 2.33 (102)

[a]The maxmium heat fluxes shown are based on reference peaking and average flux. For Cycle 1, thermal hydraulic calculations also included the densification spike factor in the DNBR calculations. B&W no longer considers this spike factor in DNBR calculations, as described in Section 2.6.

[b]140.2 inches is a conservative (minimum) value used in Cycle 2 and 3 analyses; it is the minimum densified length for any B&W fuel.

4.4 Evaluation of Thermal Hydraulic Technical Specification Changes

The staff has reviewed hot leg temperatures and low flow limits in Table 3.2-1 of the proposed technical specification for Cycle 3 operation (Reference 4-4). The same procedures and techniques were employed to derive the values in this table for this cycle as were used for previous cycles. These values are the same or slightly different from previous cycles and are consistent with the changes in the parameters. The minimum reactor coolant flow rates of 139.7 X 10^6 lbs/hr for four pump operation and 104.4 X 10^6 lbs/hr for three pump operation reflect the values used in the analyses. Therefore we conclude that these changes are acceptable.

References

4-1 Crystal River Unit 3, Cycle 3 Reload Report, BAW-1607, Rev. 1, Babcock and Wilcox, Lynchburg, Virginia, April 1980.

4-2 Correlation of Critical Heat Flux in Bundle Cooled by Pressurized Water, BAW-10000A, Babcock and Wilcox, Lynchburg, Virginia, May 1976.

4-3 Crystal River Unit 3, Cycle 2 Reload Report, BAW-1521, Babcock and Wilcox, Lynchburg, Virginia, February 1979.

4-4 Letter R. M. Bright, FPC, to Director, NRR, NRC, April 30, 1980.

4-5 Crystal River Unit 3, Fuel Densification Report, BAW-1397, Babcock and Wilcox, Lynchburg, Virginia, August 1973.

4-6 S. A. Varga (NRC) to J. H. Taylor (B&W), Letter, "Update of BAW-10055, Fuel Densification Report", December 5, 1977.

4-7 Letter R. M. Bright, FPC, to Director, NRR, NRC, April 30, 1980.

parameters discussed below is to lower the MDNBR obtained during the course of the transient.

a. Design initial power level is 102% of 2544 MWt, while the value used in the analysis is 102% of 2568 MWt.

b. The Cycle 3 flow rate is 109.5% of 352,000 gpm, while the value used in the analysis is 106.5%.

c. Core flow rate as a function of time following a LOCF event is expected to be larger than shown in the FSAR Figure 14-17 for four-pump coastdown and larger than shown in Figure 14-19a for the locked-rotor. However, the analysis used the flow rates shown in the above mentioned figures.

d. Expected values at the beginning of Cycle 3 (the worst time during the cycle life for a LOCF event) of the Doppler coefficient, the moderator temperature coefficient, and the design radial x local power peaking factor ($F_{\Delta H}$) are -1.52×10^{-5} $\Delta k/k.°F$, -0.30×10^{-4} $\Delta k/k.°F$, and 1.47 respectively. The values used for the above parameters in the LOCF analysis are -1.27×10^{-5} $\Delta k/k.°F$, 0.0 $\Delta k/k.°F$, and 1.71 respectively.

The minimum DNBR obtained during this transient is 2.10 which is well above the 1.45 FSAR value. It is noteworthy that two principal differences between the FSAR analysis and the latest Cycle 3 analysis are that the FSAR analysis used W-3 CHF correlation and a reactor protection system (RPS) flux/flow trip delay time of 1.40 sec before the control rods start to move into the core, while the Cycle 3 analysis used the BAW-2 CHF correlation and an RPS RCPPM trip delay time of 0.62 sec.

5.2.2 Locked-Rotor

The locked-rotor event is analyzed using the same conservative assumptions used in the four-pump coastdown transient discussed above. An additional assumption for the locked-rotor event was to initiate film boiling at a DNBR of 1.43 instead of the 1.3 limit.

The licensee concluded that less than 0.5% of the fuel pins in the core will experience a DNBR less than 1.43, and no pins will experience a DNBR less than 1.00. Even if the 0.5% of the fuel pins which experienced a DNBR less than 1.43 were to fail, the offsite dose releases resulting from a locked rotor event are expected to be a small fraction of 10 CFR 100 limits (see Table 5.2).

5.2.3 Conclusion

Based on the above we conclude that the accident and transient analysis is acceptable.

TABLE 5.1

Comparative Review of FSAR and Cycle 3 Parameters for Some Key Events

Event, Parameter	FSAR	Cycle 3
Rod Withdrawal		
BOL: Doppler ($\Delta k/k.°F$)	-1.17×10^{-5}	-1.52×10^{-5}
MTC ($\Delta k/k.°F$)	0.0	-0.30×10^{-4}
max. rod worth ($\%\Delta k/k$)	up to 12.9	< 9.37
Mod. Dilution		
BOL: Boron Conc. (ppm)	1150	1185
Boron worth ($ppm/\frac{\%\Delta k}{k}$)	100	108
MTC ($\Delta k/k.°F$)	$+0.5 \times 10^{-4}$	-0.3×10^{-4}
Dilution rate (gpm)	up to 500	< 100
Cold Water (2-RCP start)		
EOL: Doppler ($\Delta k/k.°F$)	-1.3×10^{-5}	-1.6×10^{-5}
MTC ($\Delta k/k.°F$)	-4.0×10^{-4}	-2.63×10^{-4}
Rod Drop		
EOL: Doppler ($\Delta k/k.°F$)	-1.3×10^{-5}	-1.61×10^{-5}
MTC ($\Delta k/k.°F$)	-3.0×10^{-4}	-2.63×10^{-4}
max. rod worth ($\% \Delta k/k$)	0.40	0.20
MSLB		
EOL: MTC ($\Delta k/k.°F$)	-3.0×10^{-4}	-2.63×10^{-4}
Ejected Rod		
BOL: Doppler ($\Delta k/k.°F$)	-1.17×10^{-5}	-1.52×10^{-5}
MTC ($\Delta k/k.°F$)	0.0	-0.3×10^{-4}
max. rod worth ($\% \Delta k/k$)	0.65	0.49
MFWLB		
BOL: Doppler ($\Delta k/k.°F$)	-1.17×10^{-5}	-1.52×10^{-5}
MTC ($\Delta k/k.°F$)	0.0	-0.3×10^{-4}

5.0 Evaluation of Accidents and Transients

5.1 General

The licensee examined each FSAR accident and transient with respect to changes in Cycle 3 parameters to determine the effect of upgrading the reactor power from 2452 to 2544 MWt. All FSAR accidents and transients with the exception of the loss-of-coolant flow (LOCF), i.e., the four-pump coastdown and the locked-rotor transients, were analyzed during the FSAR stage at 2568 MWt. This power level is higher than the requested power upgrade of 2544 MWt. Except for the LOCF, the licensee examined all FSAR accidents and transients relative to Cycle 3 operation by comparing input parameters as stated in the FSAR and as calculated for Cycle 3 and concluded that they are bounded by the FSAR and the fuel densification report analyses (References 5-1 and 5-3). The four-pump coastdown and the locked-rotor transients were reanalyzed at 102% of 2568 MWt for consistency with other B&W reactors. The LOCF event is discussed in Section 5.2 below.

A comparative review of reactivity parameters of FSAR accidents and transients except the LOCF is shown in Table 5.1. The applicability of the FSAR and reload report analyses to Cycle 3 operation is summarized in Table 5.2.

5.2 Loss-of-Coolant Flow (LOCF)

At the Cycle 2 power level of 2452 MWt, the reactor protection system depends on the flux-to-flow comparator to trip the reactor to avoid a MDNBR less than 1.3 for a 1-, 2-, 3-, or 4-pump coastdown transient. However, at the requested power upgrade of 2544 MWt, the flux-to-flow comparator, which has a trip delay time of 1.40 sec, is too slow to avoid violating the DNBR criterion for 2-, 3-, or 4-pump coastdown events. Therefore, the licensee has submitted for NRC's review and approval a proposal to add a reactor coolant pump power monitor (RCPPM)* which will continuously monitor each RCP power supply and upon power interruption to two or more RCPs will send a trip signal to the control rods with a total trip delay time of 0.62 sec. This faster trip response decreases the time during which the reactor flux-to-flow ratio exceeds the operating values and maintains the MDNBR above the 1.3 criterion during the course of the event.

5.2.1 Four-Pump Coastdown

The four-pump coastdown transient was reanalyzed at 102% of 2568 MWt assuming conservative input parameters as compared to Cycle 3 expected parameters. The effect of using the conservative

*RCPPM is being reviewed currently by NRC.

TABLE 5.2 Applicability of FSAR and Reload Report Analyses Power Level to Cycle 3

Accident/Transient	Analysis Reference	Analysis Power level, MWt	Status of Analysis Relative to Cycle 3: Percent of 2544 MWt	Status of Analysis Relative to Cycle 3: Remarks
Rod Withdrawal	FSAR	100% of 2568	101%	see footnote 1
Moderator Dilution	FSAR	100% of 2568	101%	see footnote 2
Cold Water (2-pump start)	FSAR	50% of 2568	50.5%	see footnote 3
4-PCD	Reload Report	102% of 2568	103%	Bounding
Locked-Rotor	Reload Report	102% of 2568	103%	Bounding
Stuck-in, Stuck-out, Rod Drop	FSAR	100% of 2568	101%	see footnote 4
Loss of Electrical Power	FSAR	100% of 2568	101%	see footnote 5
SLB	FSAR	100% of 2568	101%	see footnote 1
S.G. Tube Rupture	FSAR	100% of 2568	101%	see footnote 5
Fuel Handling	FSAR	100% of 2568	101%	see footnote 5
Rod Ejection	FSAR	100% of 2568	101%	see footnote 1
Max. Hypothetical Accident	FSAR	100% of 2568	101%	see footnote 5
Waste Gas Tank Rupture	FSAR	100% of 2568	101%	see footnote 5
LOCA	References 4, 5, 6	100% of 2772	109%	Bounding
MFWLB	FSAR	100% of 2568	101%	see footnote 2
Letdown Line Rupture Outside Containment	Reload Report	100% of 2603	102%	Bounding

Footnotes (Table 5.2)

1. The FSAR analysis assumed the reactor power before the accident to be 100% of 2568 MWt and the reactor is assumed to trip at 112% of 2568 MWt. This is more conservative than starting from 102% of 2544 MWT and tripping at 110% of 2544 MWt since more energy is added to the system for the FSAR analysis assumptions.

2. The FSAR analysis assumed the reactor power before the accident to be 100% of 2568. While the effect of a higher initial power of 102% of 2544 MWt (2595 MWt) is to cause the pressure trip to occur slightly sooner and the peak pressure to be slightly higher, the peak pressure is expected to be lower than the code safety limit of 2750 psia.

3. If the two pumps are started from 52% of 2544 MWt, the transient will produce a slightly higher neutron power, thermal power, and peak pressure. Since the FSAR analysis (at 50.5% of 2544 MWt) produced maximum neutron power of 75%, maximum thermal power of 65%, and a 150 psi increase over steady-state pressure of 2200 psi, the steady-state power increase is not expected to produce peak thermal power or peak pressure higher than the overpower safety limit of 112% or the code pressure limit of 2750 psia.

   FPC has been operating the Crystal River 3 plant since Cycle 2 with a modified Technical Specification that does not allow plant operation with less than three RCPs on. Therefore, the cold water accident presented in the FSAR is not directly applicable to the CR-3 Cycle 3 operation. However, an inadvertent one-pump start would decrease the RCS T_{ave} by 2°F to 3°F as compared to 7°F for two-pump start. The power and pressure surges due to a one-pump restart would be proportional to the degree of T_{ave} decrease. Therefore, a one-pump start with three-pump operation is not expected to exceed the overpower safety limit of 112% or the code pressure limit of 2750 psia.

4. Starting the transient at 102% of 2544 MWt would yield about 2350 psia peak pressure during the transient, which is much less than the code limit of 2750 psia.

5. The primary concern for this event is the radioactivity releases. The licensee has analyzed these consequences and states that they are well below the 10 CFR 100 limits.

5.3 References

5-1 CR-3, FSAR, Docket 50-302, FPC

5-2 CR-3, Cycle-3 Reload Report, BAW-1607, Rev. 1, April 1980.

5-3 CR-3, Fuel Densification Report, BAW-1397, August 1973.

5-4 ECCS Analysis of B&W's 177-FA Lowered-Loop NSS, BAW-10103A, Rev. 3, July 1977.

5-5 Letter, J. H. Taylor (B&W) to R. L. Baer (NRC), "LOCA Analysis for B&W's 177-FA Plants With Lowered Loop Arrangement (Category 1 plants) Utilizing a Revised System Pressure Distribution," July 8, 1977.

5-6 Letter, W. P. Stewart (FPC) to R. W. Reid (NRC), "Crystal River-3, Docket No. 50-302, Operating License No. DPR-72, ECCS Small Break Analysis," January 12, 1979.

6.0 Conclusions

We have evaluated the reloading of CR-3 for Cycle 3 operation and the proposed Technical Specification modifications that reflect the new cycle parameters. In the original submittal, the licensee had intended to start Cycle 3 operation at an upgraded power level of 2544 MWt. Consequently, normal operation, transients and accidents have been reanalyzed and reviewed for this increased power level. However, Cycle 3 will start at the same Cycle 2 power level of 2452 MWt.

After evaluating the FPC submittals, we conclude that CR-3 operation at or below 2452 MWt is acceptable.

We have determined that the amendment for Cycle 3 operation at 2452 MWt 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 for Cycle 3 operation at 2452 MWt. We will, however, prepare an Environmental Impact Appraisal in connection with the licensee's request to allow operation of CR-3 at increased power levels up to 2544 MWt. This document will be issued concurrently with any further Commission action concerning operation at this increased power level.

We have concluded, based on CR-3 Cycle 3 operation at 2452 MWt and the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of an 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: August 1, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-302

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

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 32 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 for operation for the Crystal River Unit No. 3 Nuclear Generating Plant (the facility) located in Citrus County, Florida. The amendment is effective as of the date of issuance.

The amendment revises the Appendix A Technical Specifications to permit power operation during Cycle 3 at the currently authorized power level of 2452 MWt.

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 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 March 21, 1980, as revised and supplemented April 14, 1980, April 30, 1980, June 6, 1980, June 13, 1980, July 22, 1980, and July 31, 1980, (2) Amendment No. 32 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, 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 1st day of August 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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