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JULY 1979

Docket No.: 50-302

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Mr. W. P. Stewart
Director, Power Production
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Stewart:

The Commission has issued the enclosed Amendment No. 19 to Facility Operating License No. DPR-72 for Crystal River Unit No. 3 Nuclear Generating Plant. This amendment consists of changes to the Technical Specifications in partial response to your applications dated February 28, 1979 and March 15, 1979, as supplemented on May 25, 1979.

This amendment revises the Appendix A Technical Specifications to permit power operation during Cycle 2 at the previously authorized power level of 2452 MWt.

Your application of March 15, 1979, proposed Cycle 2 operation at 2544 MWt and was subsequently amended to request Cycle 2 restart at 2452 MWt with the power increase to occur later in Cycle 2. The enclosed Safety Evaluation documents our review of the reloaded reactor core and accidents and transients for operation at 2452 MWt and 2544 MWt. It does not address the radiological consequences of an accident at 2544 MWt nor the nonradiological environmental impact of operation at the higher power level. Prior to further Commission action regarding your application for the power increase, you must propose those Technical Specification changes necessary to reflect partial Cycle 2 operation at 2452 MWt.

~~Don't we mean 2544?~~ No.

As requested in your letter of March 14, 1979, as supplemented on June 5, 1979, the Commission has issued the enclosed Exemption for Crystal River Unit No. 3 from the requirements of 10 CFR 50.46(a)(1).

7908030304

CP 1

60

OFFICE						
SURNAME						
DATE						

On March 28, 1979, Three Mile Island Unit No. 2 (TMI-2) experienced core damage which resulted from a series of events which were initiated by a loss of feedwater and apparently compounded by operational errors. Through IE Bulletins 79-05, 05A and 05B dated, respectively, April 1, 5 and 21, 1979, issued to licensees of Babcock & Wilcox reactors, the NRC Office of Inspection and Enforcement identified corrective actions to be taken at Crystal River Unit No. 3.

Why are we discussing this here if it is being addressed in the separate package

You responded to this group of Bulletins by letters dated April 9, 12 and 22, May 4 and 21, and June 15, 1979. Our preliminary evaluation of your responses and actions finds that you have demonstrated understanding of the TMI-2 event and its relationship to Crystal River Unit No. 3. Your actions provide added protection to the health and safety of the public during plant operation. A separate Safety Evaluation will be issued to document our review of your Bulletin responses. We may also request additional information and identify required future actions relevant to your Bulletin responses.

You did, by letter dated May 1, 1979, commit to perform certain actions prior to Cycle 2 restart as a result of review of the TMI-2 experience. The need to perform these actions was confirmed by the Commission in its Order of May 16, 1979. The Order requires that Crystal River Unit No. 3 be maintained in a shutdown condition until satisfactory completion of the items in the Order have been confirmed by the Director, Office of Nuclear Reactor Regulation. Therefore, upon issuance of this amendment related to the fuel loading for Cycle 2, operation of the facility can be commenced only after confirmation of completion of the items in the Order.

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.

The Exemption and the Notice of Issuance are being filed with the Office of the Federal Register for publication.

Sincerely,

Original signed by

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

Enclosures:
1. Amendment No. 19
2. Safety Evaluation
3. Exemption
4. Notice

STS
D Brinkman
6/29/79

cc w/enclosures: See next page

OFFICE	ORB#4:DOR	ORB#4:DOR	C-ORB#4:DOR	A-AD:ORP:DOR	OELD	
SURNAME	RIngram	CNelson:rf	RReid	WGammill	S H Lewis	
DATE	6/21/79	6/27/79	6/27/79	6/28/79	6/29/79	

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

July 3, 1979

Docket No.: 50-302

Mr. W. P. Stewart
Director, Power Production
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

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7908030304

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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.

The Exemption and the Notice of Issuance are being filed with the Office of the Federal Register for publication.

Sincerely,

Robert W. Reid

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

Enclosures:
1. Amendment No. 19
2. Safety Evaluation
3. Exemption
4. Notice

cc w/enclosures: See next page

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

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Atlanta, Georgia 30308

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
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cc w/enclosures & incoming 3/14 & 6/5/79,
dtd: 2/28, 3/15 & 5/25/79
Bureau of Intergovernmental Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

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. 19
License No. DPR-72

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

   A. The applications for amendment by Florida Power Corporation, et al (the licensees) dated February 28, 1979 and March 15, 1979, as supplemented May 25, 1979, comply 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 applications, 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.

7908030307

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

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.19 , 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 for

William P. Gammill, Acting Assistant Director
for Operating Reactor Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 3, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 19

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

2-2
2-3
2-5
B 2-1
B 2-6
B 2-7
B 2-8 (added)
3/4 1-27
3/4 1-28
3/4 1-29
3/4 1-30
3/4 1-33
3/4 1-34
3/4 1-35
3/4 1-36
3/4 1-37
3/4 1-38
3/4 1-39 (added)
3/4 2-2
3/4 2-3
3/4 2-4
3/4 2-11
3/4 2-13
B 3/4 2-2
B 3/4 2-3

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

REACTOR CORE

2.1.1 The combination of the reactor coolant core outlet pressure and outlet temperature shall not exceed the safety limit shown in Figure 2.1-1.

APPLICABILITY: MODES 1 and 2.

ACTION:

Whenever the point defined by the combination of reactor coolant core outlet pressure and outlet temperature has exceeded the safety limit, be in HOT STANDBY within one hour.

REACTOR CORE

2.1.2 The combination of reactor THERMAL POWER and AXIAL POWER IMBALANCE shall not exceed the safety limit shown in Figure 2.1-2 for the various combinations of three and four reactor coolant pump operation.

APPLICABILITY: MODE 1.

ACTION:

Whenever the point defined by the combination of Reactor Coolant System flow, AXIAL POWER IMBALANCE and THERMAL POWER has exceeded the appropriate safety limit, be in HOT STANDBY within one hour.

REACTOR COOLANT SYSTEM PRESSURE

2.1.3 The Reactor Coolant System pressure shall not exceed 2750 psig.

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

ACTION:

MODES 1 and 2 Whenever the Reactor Coolant System pressure has exceeded 2750 psig, be in HOT STANDBY with the Reactor Coolant System pressure within its limit within one hour.

MODES 3, 4 and 5 - Whenever the Reactor Coolant System pressure has exceeded 2750 psig, reduce the Reactor Coolant System pressure to within its limit within 5 minutes.

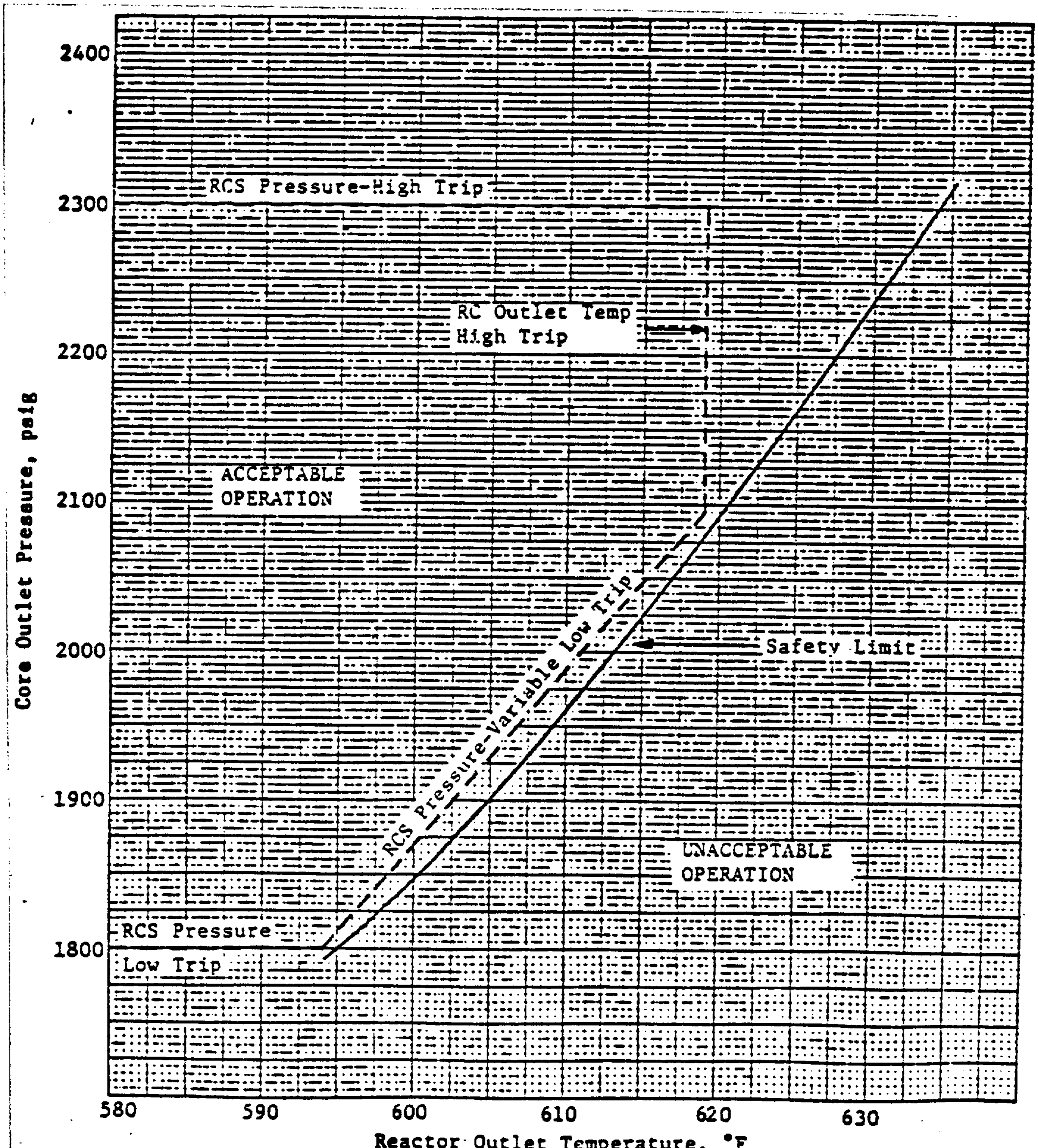
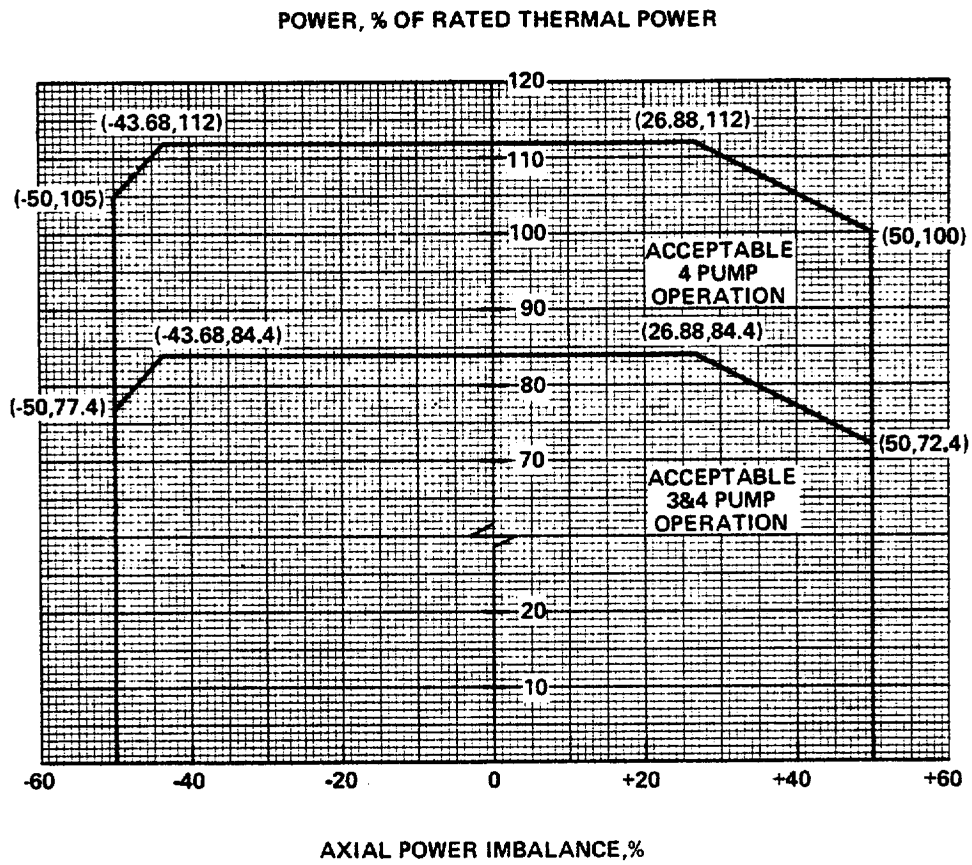


FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT



CURVE	REACTOR COOLANT FLOW (LB/HR)
1	139.86×10^6
2	104.47×10^6

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	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating $\leq$ 78% of RATED THERMAL POWER with three pumps operating	$\leq$ 105.5% of RATED THERMAL POWER with four pumps operating $\leq$ 78% of RATED THERMAL POWER with three pumps operating
3. RCS Outlet Temperature-High	$\leq$ 619°F	$\leq$ 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)	$\geq$ 1800 psig	$\geq$ 1800 psig
6. RCS Pressure-High	$\leq$ 2300 psig	$\leq$ 2300 psig
7. RCS Pressure-Variable Low(1)	$\geq$ (11.80 T_{out} °F - 5209.2) psig	$\geq$ (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 $\leq$ 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.

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 139.86 x 10^6 lbs/hr, which is 106.5% of the design flow rate for four operating reactor coolant pumps. This curve is based on the following nuclear power peaking factors with potential fuel densification effects:

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

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 or the local quality at the point of minimum DNBR is equal to 22%, whichever condition is more restrictive.

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.

LIMITING SAFETY SYSTEM SETTINGS

BASES

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 $\geq$ 104.3% and reactor flow rate is 100%, or flow rate is $\leq$ 95.9% and power level is 100%.

2. Trip would occur when three reactor coolant pumps are operating if power is $\geq$ 77.9% and reactor flow rate is 74.7%, or flow rate is $\leq$ 71.9% and power is 75%.

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

LIMITING SAFETY SYSTEM SETTINGS

BASES

The AXIAL POWER IMBALANCE boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The AXIAL POWER IMBALANCE reduces the power level trip produced by the flux-to-flow ratio such that the boundaries of Figure 2.2-1 are produced. The flux-to-flow ratio reduces the power level trip and associated reactor power-reactor power-imbalance boundaries by 1.043% 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.

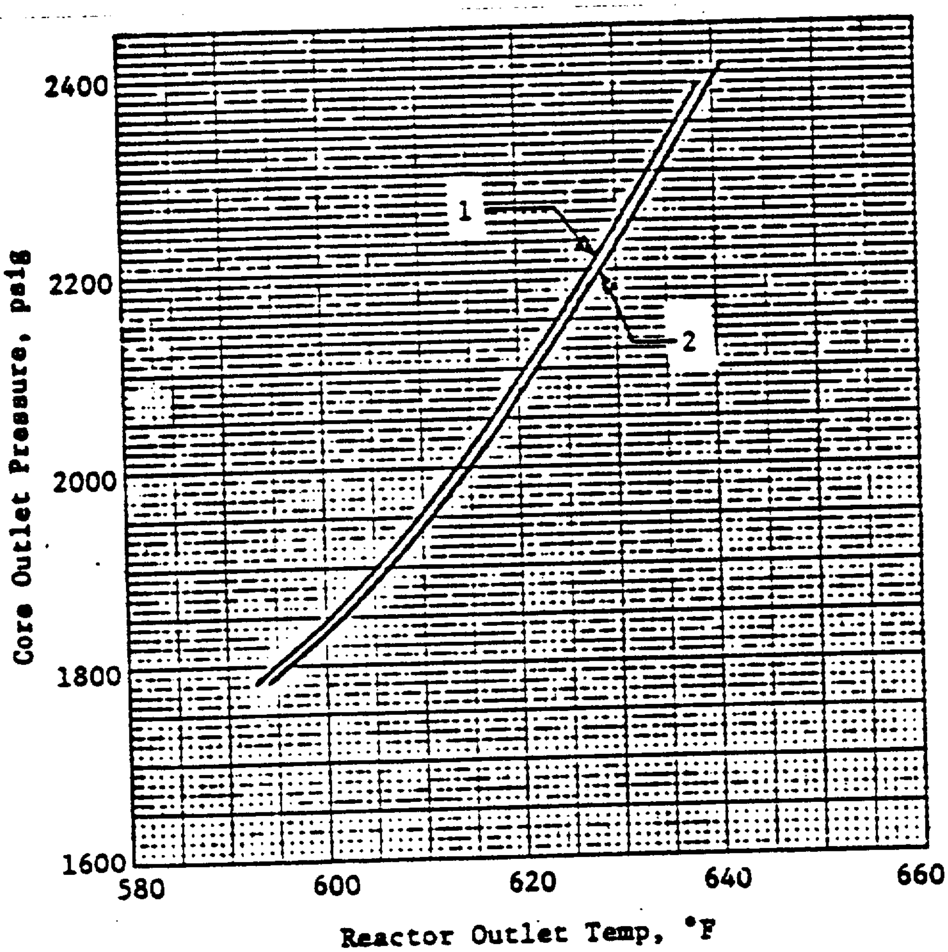
Due to the calibration and instrumentation errors, the safety analysis used a RCS Pressure-Variable Low Trip Setpoint of (11.80 T_{out}°F-5249.2) psig.

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.



REACTOR COOLANT FLOW

CURVE	FLOW (lb/hr)	POWER(%RTP)	PUMPS OPERATING (TYPE OF LIMIT)
1	139.86×10^6 (106.7%)	117.3%	4 Pumps (DNBR)
2	104.47×10^6 (79.7%)	90.5%	3 Pumps (DNBR)

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM
ALLOWABLE POWER FOR MINIMUM DNBR

BASES FIGURE 2.1

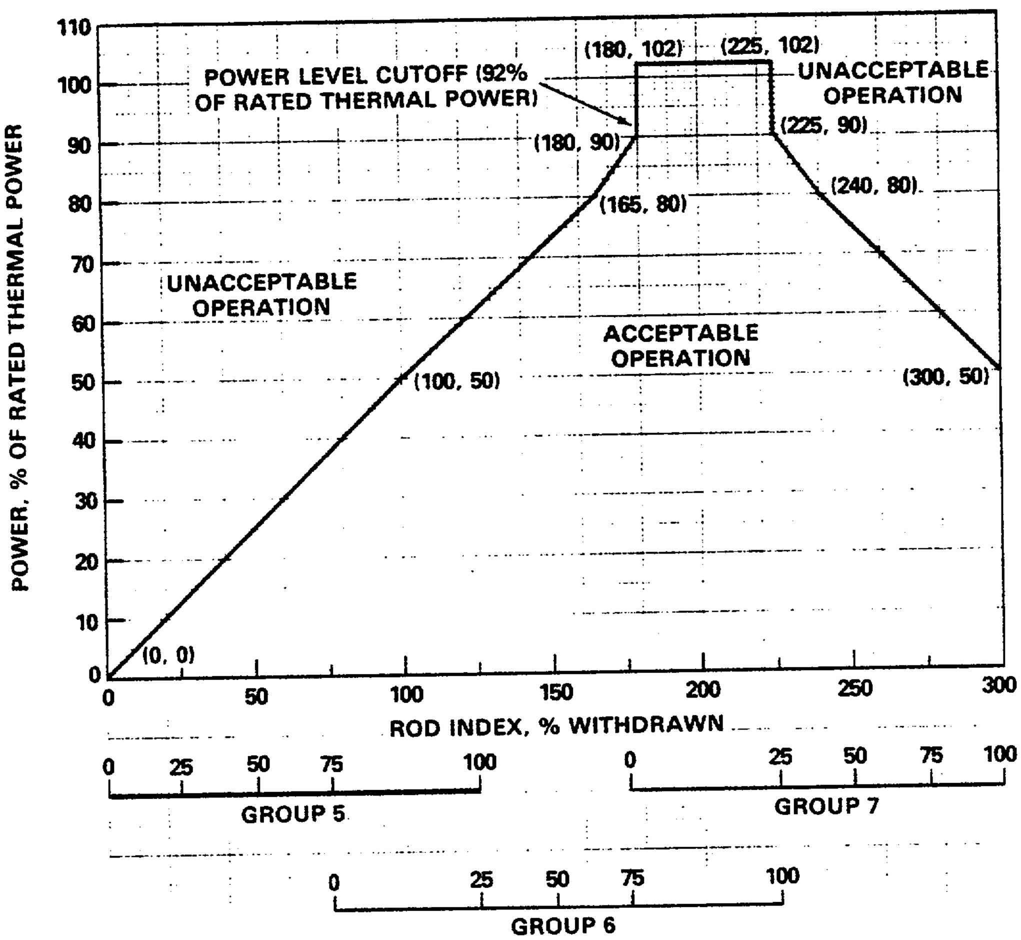


Figure 3.1-1
Regulating Rod Group Insertion Limits For 4 Pump Operation From 0 EFPD To 233 ± 10 EFPD

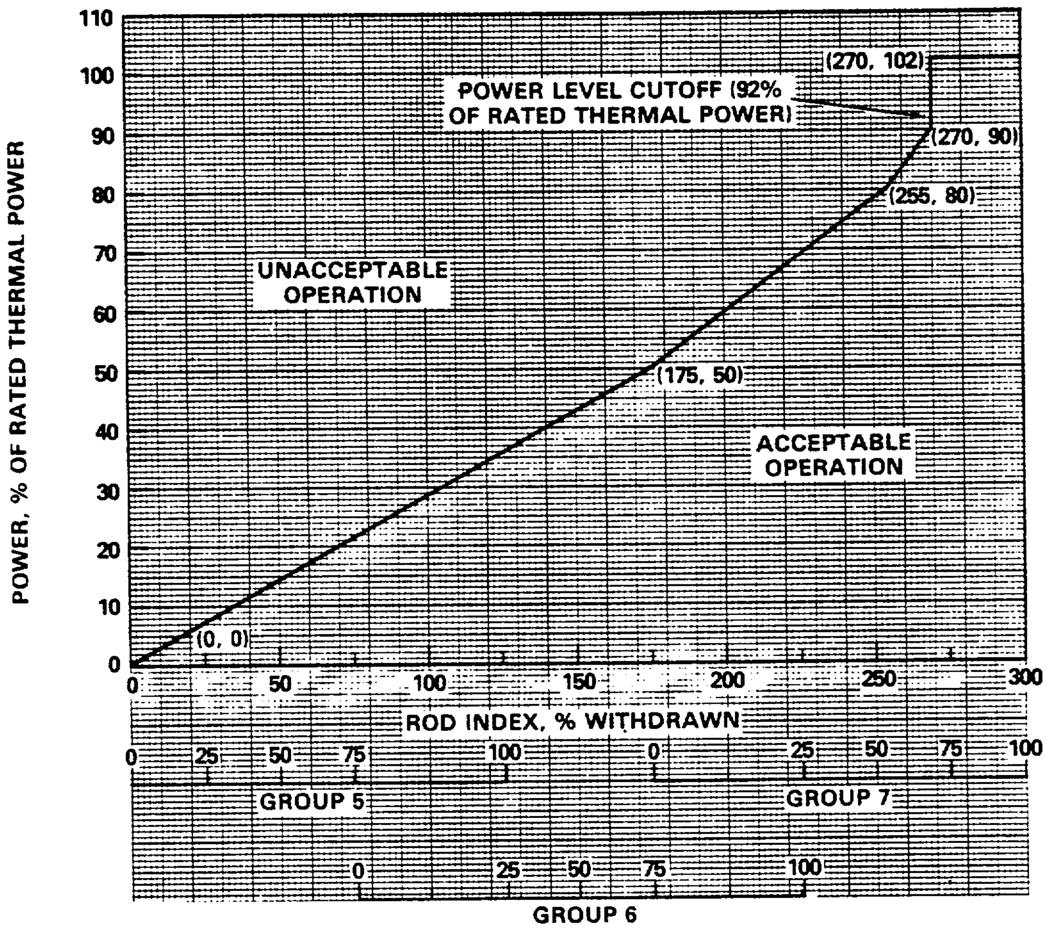


Figure 3.1-2
Regulating Rod Group Insertion Limits For 4 Pump Operation After 233 ± 10 EFPD

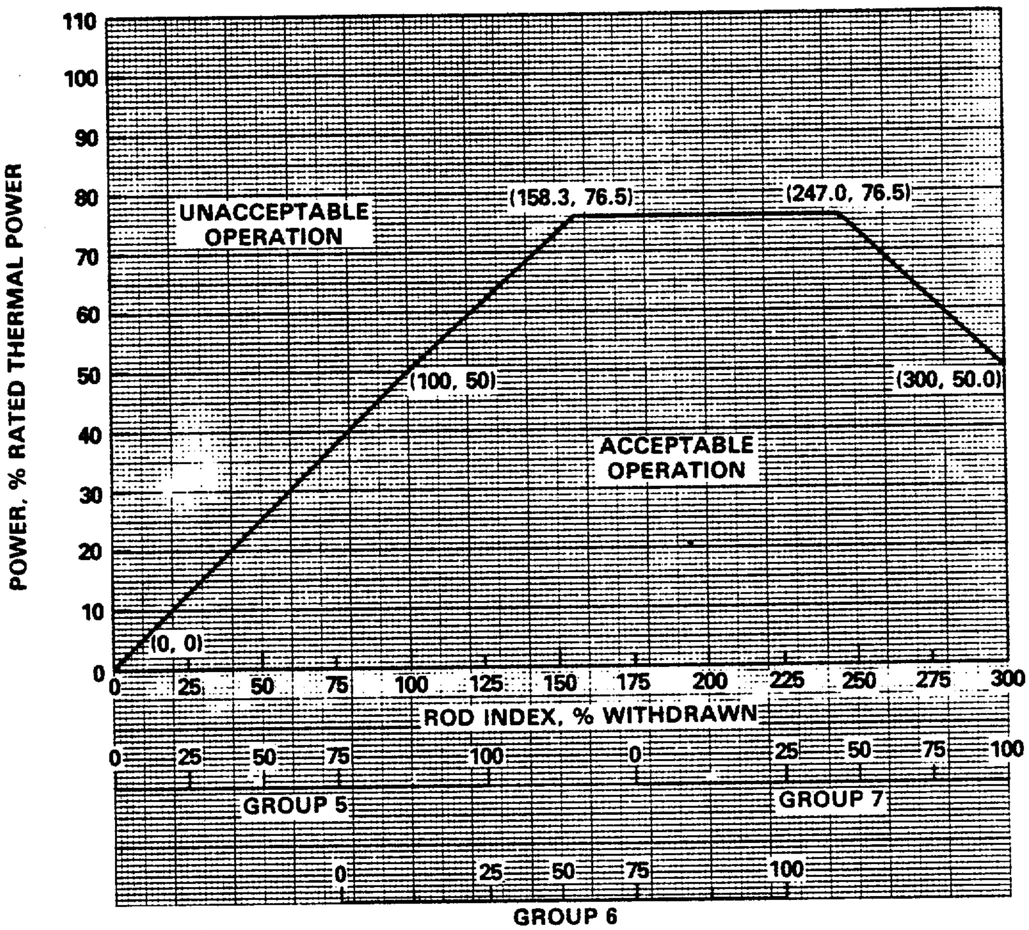


Figure 3.1-3
Regulating Rod Group Insertion Limits For 3 Pump Operation From 0 EFPD To 233 ± 10 EFPD

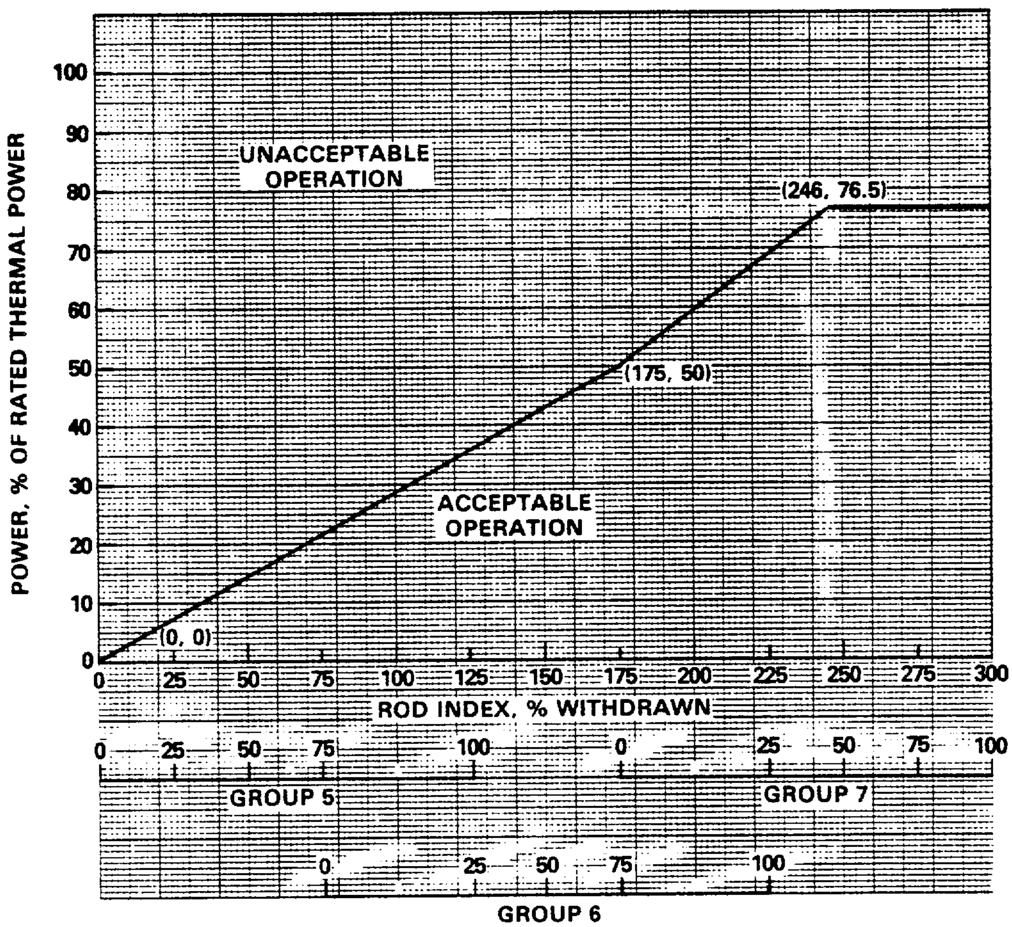


Figure 3.1-4
Regulating Rod Group Insertion Limits For 3 Pump Operation After 233 ± 10 EFPD

REACTIVITY CONTROL SYSTEMS

ROD PROGRAM

LIMITING CONDITION FOR OPERATION

3.1.3.7 Each control rod (safety, regulating and APSR) shall be programmed to operate in the core position and rod group specified in Figure 3.1-7.

APPLICABILITY: MODES 1* and 2*.

ACTION:

With any control rod not programmed to operate as specified above, be in HOT STANDBY within 1 hour.

SURVEILLANCE REQUIREMENTS

4.1.3.7

a. Each control rod shall be demonstrated to be programmed to operate in the specified core position and rod group by:

   1. Selection and actuation from the control room and verification of movement of the proper rod as indicated by both the absolute and relative position indicators:

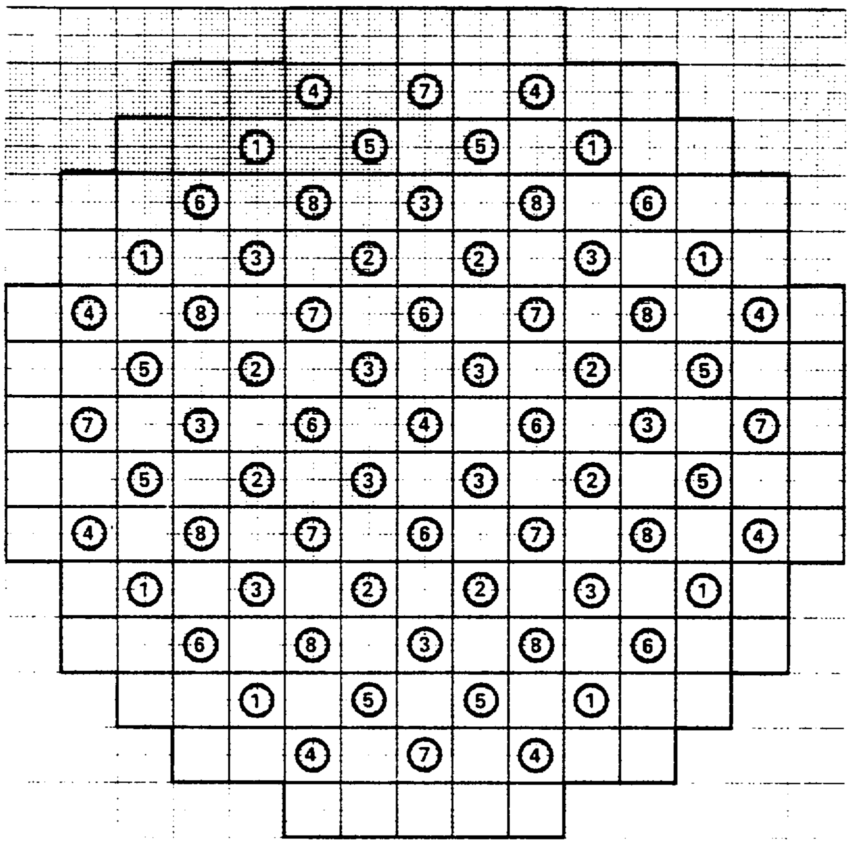
      a) For all control rods, after the control rod drive patchs are locked subsequent to test, reprogramming or maintenance within the panels.

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

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

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

*See Special Test Exceptions 3.10.1 and 3.10.2.



GROUP	NUMBER OF RODS	FUNCTION
1	8	SAFETY
2	8	SAFETY
3	12	SAFETY
4	9	SAFETY
5	8	CONTROL
6	8	CONTROL
7	8	CONTROL
8	8	APSRs
TOTAL	69	

Figure 3.1-7
Control Rod Locations And Group Assignments

DELETED

XENON REACTIVITY

LIMITING CONDITION FOR OPERATION

3.1.3.8 THERMAL POWER shall not be increased above the power level cutoff specified in Figures 3.1-1 and 3.1-2 unless xenon reactivity is within 10 percent of the equilibrium value for RATED THERMAL POWER and is approaching stability.

APPLICABILITY: MODE 1.

ACTION:

With the requirements of the above specification not satisfied, reduce THERMAL POWER to less than or equal to the power level cutoff within 15 minutes.

SURVEILLANCE REQUIREMENTS

4.1.3.8 Xenon reactivity shall be determined to be within 10% of the equilibrium value for RATED THERMAL POWER and to be approaching stability prior to increasing THERMAL POWER above the power level cutoff.

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 $k_{eff} \geq 1.0$.

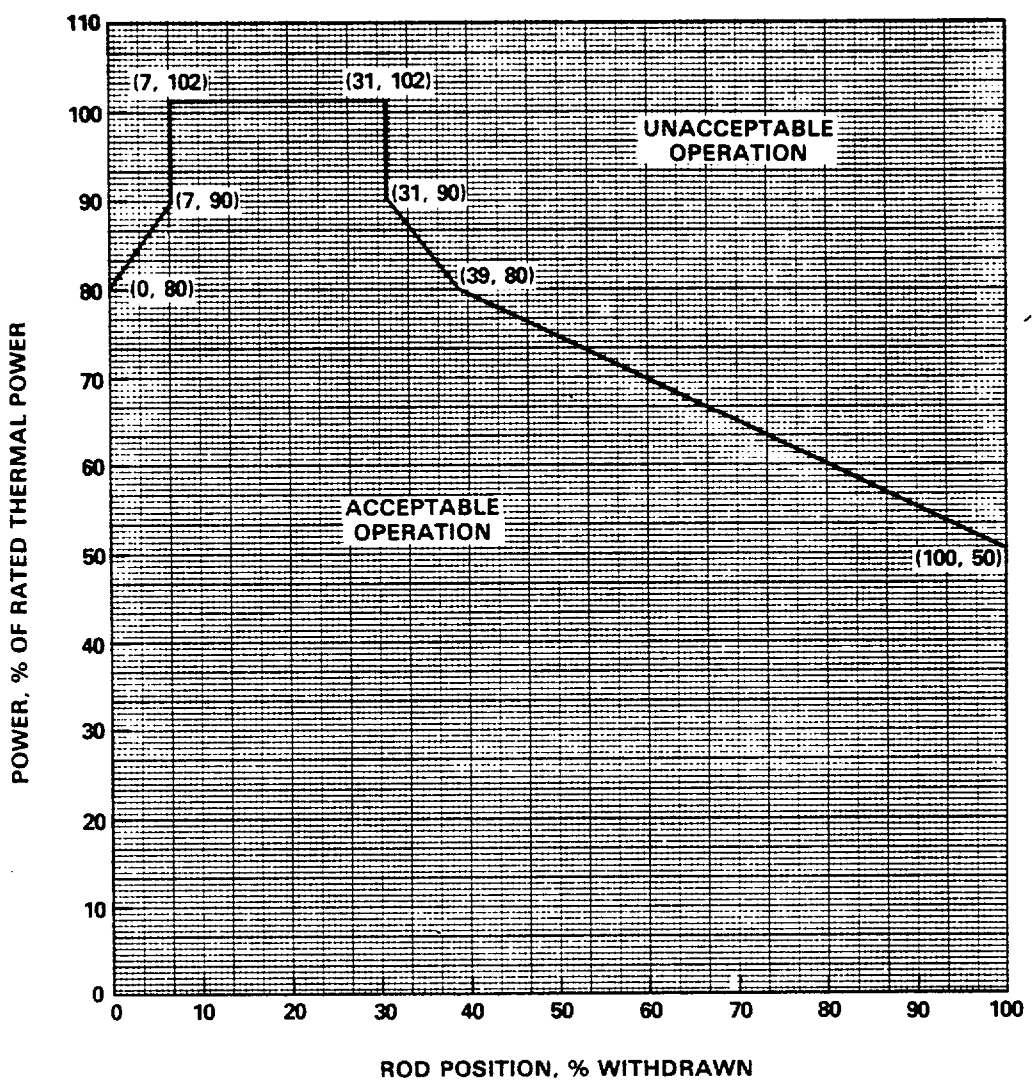


Figure 3.1-9
Axial Power Shaping Rod Group Insertion Limits
From 0 EFPD To 233 ± 10 EFPD

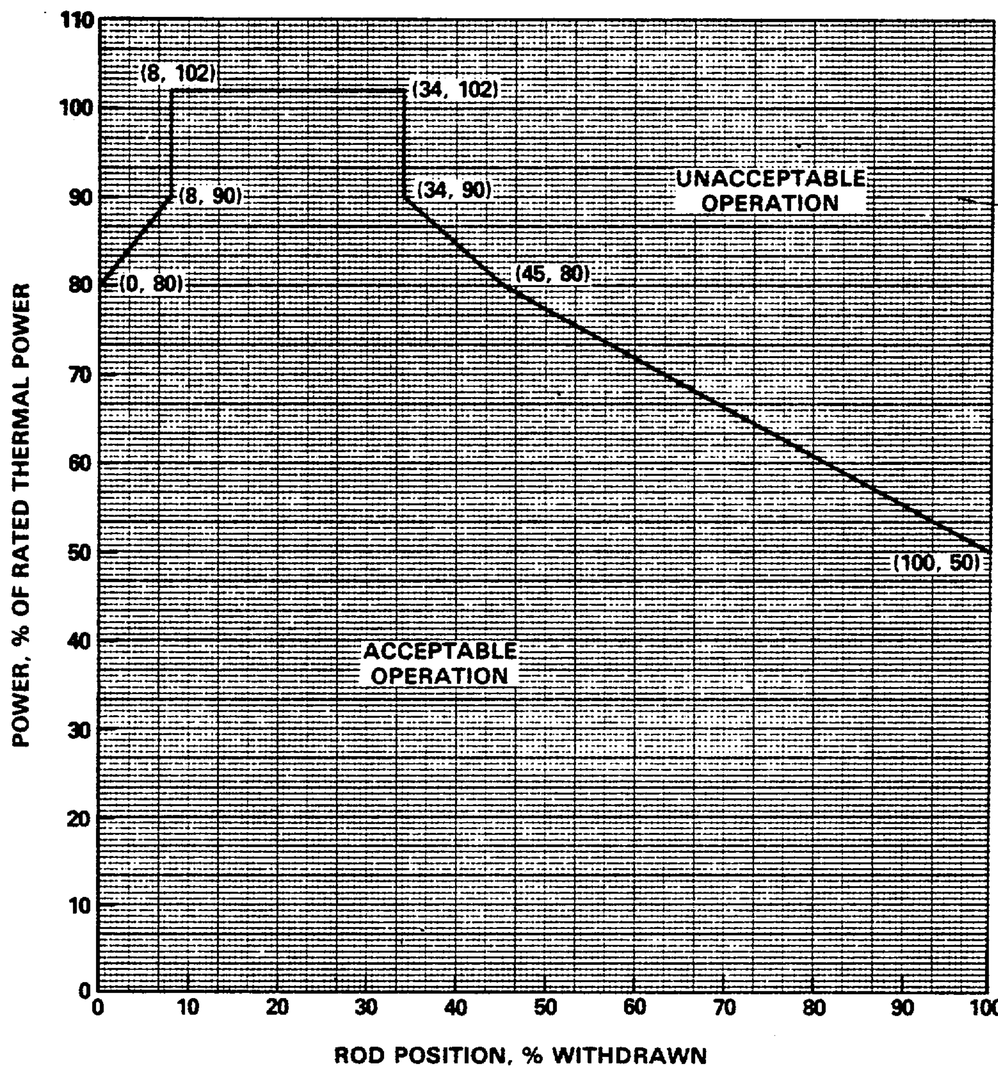


Figure 3.1-10
Axial Power Shaping Rod Group
Insertion Limits After 233 ± 10 EFPD

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.

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*See Special Test Exception 3.10.1.

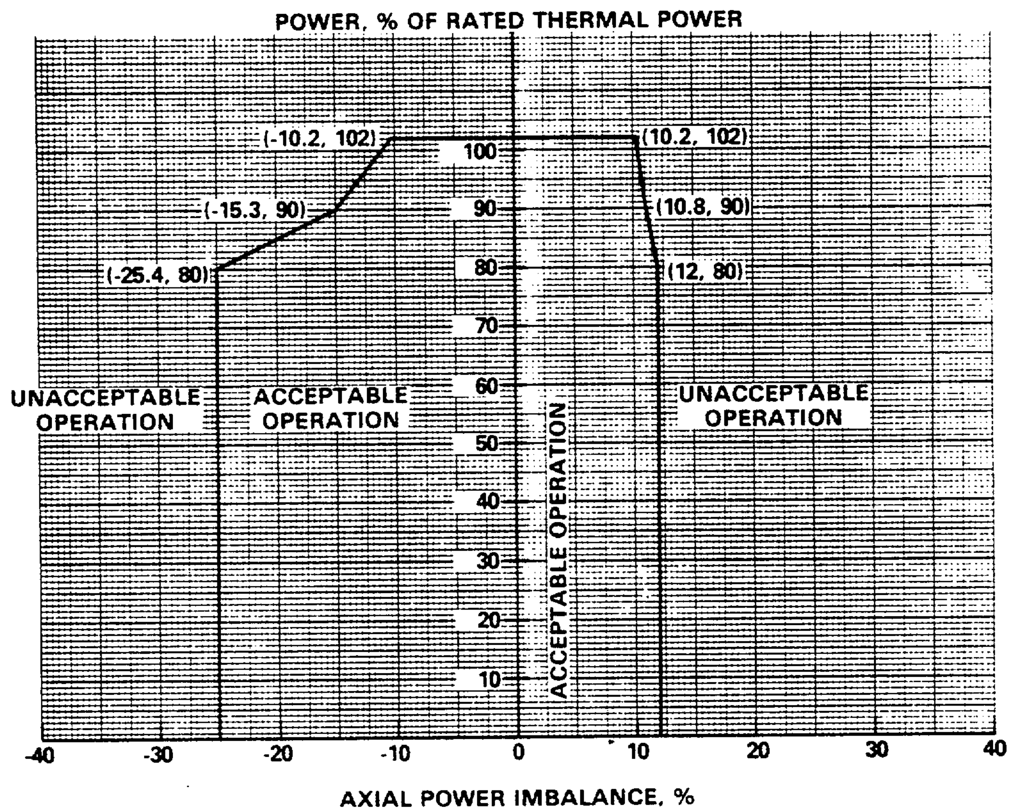

Figure 3.2-1
Axial Power Imbalance Envelope For
Operation From 0 EFPD To 233 ± 10 EFPD

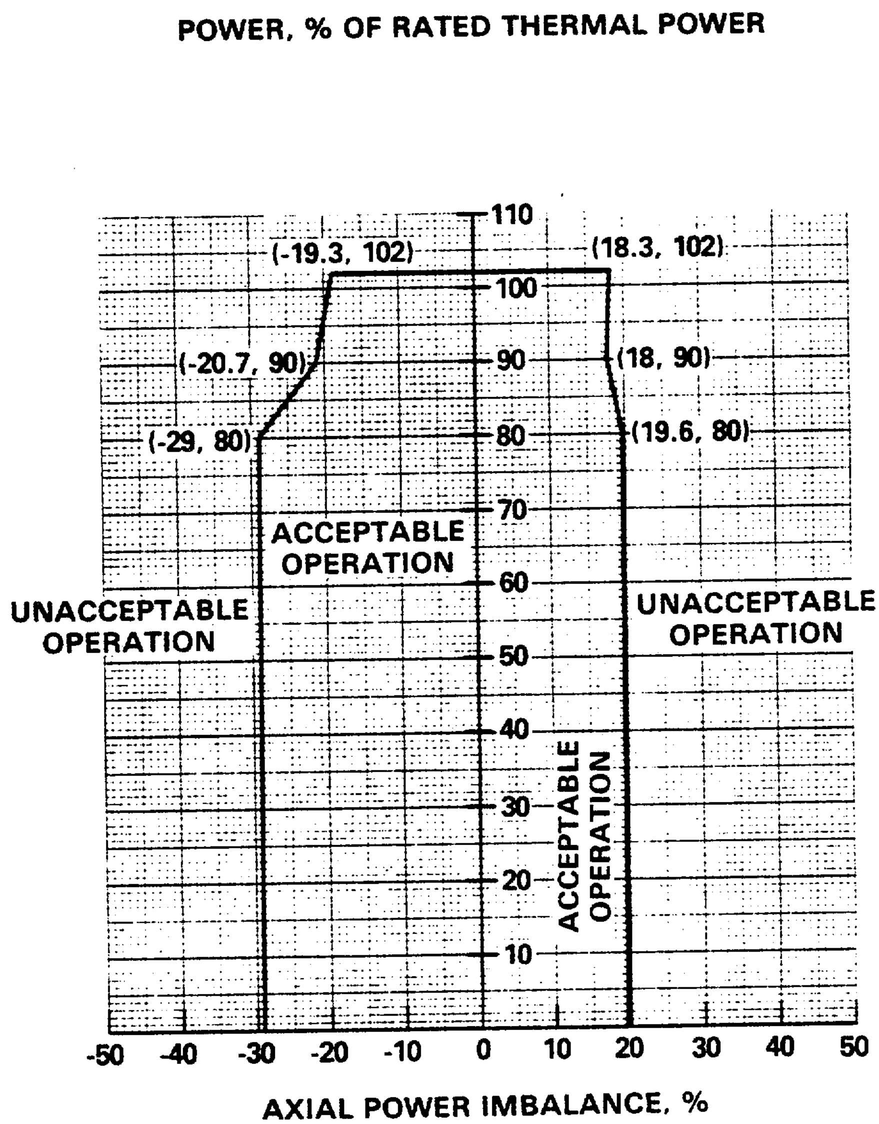

Figure 3.2-2
Axial Power Imbalance Envelope For Operation After 233 ± 10 EFPD

POWER DISTRIBUTION LIMITS

NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

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

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

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	3.46	8.96	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	LIMITS: 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	$\geq 139.86 \times 10^6$	$\geq 104.47 \times 10^6$

(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.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core ≥ 1.30 during normal operation and during short term transients, (b) maintaining the peak linear power density ≤ 18.0 kw/ft during normal operation, and (c) maintaining the peak power density ≤ 19.7 kw/ft during short term transients. In addition, the above criteria must be met in order to meet the assumptions used for the loss-of-coolant accidents.

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

a. Nuclear uncertainty factors.

b. Thermal calibration uncertainty.

c. Fuel densification effects.

d. Hot rod manufacturing tolerance factors.

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

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

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

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

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

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

$$F_Q \leq 3.08; \quad F^N_{\Delta H} \leq 1.71$$

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

1. Control rods in a single group move together with no individual rod insertion differing by more than $\pm$ 6.5% (indicated position) from the group average height.
2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.
3. The regulating rod insertion limits of Specification 3.1.3.6 and the axial power shaping rod insertion limits of Specification 3.1.3.9 are maintained.
4. AXIAL POWER IMBALANCE limits are maintained. The AXIAL POWER IMBALANCE is a measure of the difference in power between the top and bottom halves of the core. Calculations of core average axial peaking factors for many plants and measurements from operating plants under a variety of operating conditions have been correlated with AXIAL POWER IMBALANCE. The correlation shows that the design power shape is not exceeded if the AXIAL POWER IMBALANCE is maintained within the limits of Figures 3.2-1 and 3.2-2.

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

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

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

For Condition II events, the core is protected from exceeding 19.7 kw/ft locally, and from going below a minimum DNBR of 1.30 by automatic protection on power, AXIAL POWER IMBALANCE, pressure and temperature. Only conditions 1 through 3, above, are mandatory since the AXIAL POWER IMBALANCE is an explicit input to the Reactor Protection System.

The QUADRANT POWER TILT limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation. For QUADRANT POWER TILT, the safety (measurement independent) limit for Steady State is 4.92, for Transient State is 11.07, and for the Maximum Limit is 20.0.

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

3/4.2.5 DNB PARAMETERS

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 Introduction

By letters dated February 28, 1979 and March 15, 1979 (References 1 and 2, respectively) Florida Power Corporation (FPC or the licensee) requested amendment of Appendix A to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 (CR-3) Nuclear Generating Plant.

The FPC submittal of February 28, 1979 included a Babcock & Wilcox (B&W) topical report, BAW-1521, February 1979 to support the CR-3 Cycle 2 reload and stretch power following the current Cycle 1. The topical report describes the fuel system design, nuclear design, thermal-hydraulic design, accident analyses, and startup test program. The topical report analyses support upgrading the CR-3 rated power from the current Cycle 1 2452 MWt to 2544 MWt in Cycle 2. The 2544 MWt was the ultimate core power considered in the Final Safety Analysis Report (FSAR). The design length of Cycle 2 is 275 effective full power days (EFPD) at the higher power level.

By letter dated May 25, 1979 (Reference 12), FPC amended References 1 and 2. In that submittal (Reference 12), FPC informed the NRC that they changed their plans and will resume operation in Cycle 2 with a maximum rated thermal power of 2452 MWt, the same rated power during Cycle 1. In Reference 12, the licensee also submitted Technical Specification changes that reflect their new plans. In the matter of another related item the NRC was verbally informed that the licensee will not install the reactor coolant pump power monitors (RCPPMs) during this refueling outage. Even though no power increase will be implemented when the plant resumes Cycle 2 operation, this safety review and accident analyses evaluation is based on the higher power level of 2544 MWt. Additional changes to the Technical Specifications in order to reflect the power increase and the addition of the RCPPMs will need to be considered at a later date.

At the end of Cycle 1, 56 batch 1 fuel assemblies will be discharged. Once burned fuel assembly batches 2 and 3 will be shuffled to new locations. Fresh batch 4 fuel assemblies will be loaded in the core periphery. Batch 4 consists of 56 Mark B4 fuel assemblies, while batches 2 and 3 consist of 61 and 60 Mark B3 fuel assemblies, respectively. No control rod interchanges or burnable poison rods are required for Cycle 2.

In support of the power upgrade, reactor coolant system (RCS) stresses were reviewed. Based on that review, FPC has determined that no hardware changes were required as a result of the power upgrade. For protection against the loss of flow transient at the Cycle 1 power level of 2452 MWt, the reactor protection system (RPS) action initiated by the flux-flow comparator is adequate to preclude the minimum departure from nucleate boiling ratio (DNBR) from going below 1.3 for the four-pump coastdown transient and below 1.0 for the locked rotor transient. However, at the higher Cycle 2 power level of 2544 MWt the RPS action initiated by the flux-flow comparator is not fast enough to protect against more than one pump coastdown. Therefore, FPC will install RCPPMs before implementing the higher power level. RCPPM addition will reduce the RPS response time for the above transients from 1.40 seconds to 0.62 seconds (Reference 3) for the loss of more than one reactor coolant pump (RCP) thus satisfying the departure from nucleate boiling (DNB) criteria. When the reactor power is upgraded and the RCPPMs are installed, the flux-flow comparator will protect the core against the loss of one RCP, thus providing high flux and/or low flow trips. The proposed RCPPMs are currently under review.

2.0 Evaluation of Modifications to Core Design

2.1 Fuel System Design

The fresh 56 Mark B4 fuel assemblies loaded as batch 4 at the end of Cycle 1 (EOC 1) are mechanically interchangeable with batches 2 and 3 fuel assemblies (Mark B3). Mark B4 fuel design has been previously approved and utilized at other B&W nuclear steam supply systems. The new fuel assemblies have modified end fittings, mainly to reduce fuel assembly pressure drop. The new fuel assemblies also incorporate minor design modifications to the spacer grid corner cells which reduce spacer/grid interaction during fuel handling.

2.1.1 Cladding Creep Collapse

For the cladding creep collapse analysis, batches 2 and 3 are more limiting than batch 4 due to their longer 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 (Reference 4). The analysis conservatively determined a creep collapse time of 30,000 effective full power hours (EFPH). Since this collapse time is greater than the accumulated actual exposure for the most limiting assembly at EOC 2 (Table 4-1 of BAW-1521 attached to Reference 1), we conclude that cladding creep collapse has been adequately considered.

2.1.2 Cladding Stress and Strain

Batches 2 and 3 fuel assemblies are the limiting batches relative to cladding stress due to their lower initial density and longer accumulated exposure time. Batches 2 and 3 have been analyzed and documented in Reference 5.

Fuel design criteria specify a 1% limit on cladding plastic circumferential strain. The pellet design is established for cladding plastic strain less than 1% at values of maximum design pellet burnup and heat generation rate. Those maximum design values are considerably higher than those expected for Cycle 2 operation.

2.1.3 Fuel Thermal Design

Reference 1 states that linear heat rates (LHR) are based on centerline fuel melt and were established using the TAFY-3 Code (Reference 6). Batch 4 fuel has a higher initial density than batches 2 and 3, and a correspondingly higher LHR capability (20.15 vs. 19.70 kw/ft - Table 4-2 of Reference 1). Pellet resinter test data from batch 4 will be evaluated to demonstrate that the fuel exceeds the design minimum LHR capability. The licensee has confirmed that the resinter test model and evaluation will conform with those accepted by the NRC for B&W fuel designs (Reference 13).

Densification power spike analysis for Cycle 2 is identical to that presented in BAW-10055 (Reference 7) except for modifications to F_g and F_k as described in Reference 8 and approved in Reference 9. These same modifications to the power spike model have been approved for other B&W plants.

Based on the above information, we conclude that the fuel thermal design has been adequately considered.

2.2 Nuclear Design

Core loading diagram for Cycle 2 is shown in Figure 3-1 of Reference 1. Figure 3-2 of the same reference shows the initial enrichments and burnup distributions for the beginning of Cycle 2 (BOC 2). Cycle 2 will have a projected length of 275 EFPD at a power level of 2544 MWt and a cycle burnup of 8,500 MWD/MTU. The nuclear parameters for Cycle 2 are calculated using the approved PDQ07 Code (Reference 10). These parameters are compared to those of Cycle 1 in Table 5-1 of Reference 1. Since the core has not yet reached an equilibrium cycle, differences in core nuclear parameters are expected between cycles. However, the shorter Cycle 2 will produce a smaller cycle differential burnup than Cycle 1. (Burnup at EOC 1 is 12,849 MWD/MTU, while design burnup of Cycle 2 is 8,500 MWD/MTU.)

Table 5-2 of Reference 1 shows the shutdown margins for BOC and EOC. The calculated minimum shutdown margin during Cycle 2 is 2.1%Δk/k which is larger than the value of 1%Δk/k assumed in cooldown accident analyses by an adequate margin.

2.3 Thermal Hydraulic Design

A comparison between the thermal hydraulic design conditions of Cycle 1 and Cycle 2 is listed in Table 6-1 of Reference 1. The thermal hydraulic design calculations in support of Cycle 2 operation assumed a rated power level of 2568 MWt for consistency with other B&W plants. The differences between Cycles 1 and 2 are discussed below.

2.3.1 Reactor Coolant Flow

The assumed system flow was changed from 105% (Cycle 1) to 106.5% (Cycle 2) of the design flow (88,000 gpm/pump) for consistency with other B&W plants with the same design and power level (e.g., Oconee 1, 2, 3, ANO-1, and TMI-1). This higher assumed flow rate is supported by measurements conducted by the utility at CR-3.
Those measurements indicate a system flow capability of 109.5% of design flow rate.

2.3.2 Mark B4 Fuel Assemblies

The main difference between the fresh Mark B4 fuel assemblies loaded for Cycle 2 operation and Mark B3 fuel assemblies is in the end fittings, which have been modified to reduce assembly pressure drop. The reduced pressure drop causes a slight increase in flow through the B4 assemblies relative to the B3 design. For Cycle 2 operation, the highest steady-state heat generation rate will occur in the fresh batch 4, Mark B4 fuel assemblies. However, no credit was taken for any increase in B4 assemblies' flow. Mark B4 assemblies are currently in all B&W operating reactors.

2.3.3 Fuel Rod Bow DNBR Penalty

B&W submitted an interim rod bow penalty evaluation procedure (Reference 11) until a topical report addressing this subject is completed and reviewed. In Reference 11, B&W asserts that for B&W fuel design there is no DNBR penalty due to fuel rod bow for fuel with less than approximately 21,300 MWD/MTU burnup. For CR-3, the limiting fuel rod will have a burnup of less than 21,300 MWD/MTU at the EOC 2. Therefore, B&W asserts no DNBR rod bow penalty is required. Even though the B&W's interim submittal has not been fully reviewed, there is a sufficient DNBR margin inherent in plant setpoints and limits to more than compensate for any potential revision that we may require to B&W's rod bow model. On that basis, we find the use of the referenced rod bow model acceptable.

3.0 Evaluation of Accidents and Transients

3.1 General

All FSAR accidents and transients with the exception of two, namely the four-pump coastdown and the locked-rotor transients, were analyzed assuming a thermal power level of 2568 MWt which is higher than the upgraded power level of 2544 MWt. Except for the above two transients, the licensee examined all accidents and transients analyzed in the FSAR and concluded that they are bounded by the FSAR and/or the fuel densification reports (References 5, 7, 9). The four-pump coastdown and the locked-rotor transients were reanalyzed at 102% of 2568 MWt for consistency with other B&W reactors. The applicability of the FSAR analyses to Cycle 2 is summarized in Table 1.

3.2 Four-Pump Coastdown

The four-pump coastdown transient has been analyzed assuming conservative analysis parameters. Those analysis parameters are compared to the expected parameters of Cycle 2 as follows:

Initial flow rate is 109.5% of 352,000 gpm, while the value used in analysis is only 106.5%.

Design initial power level is 102% of 2544 MWt, while the value used in analysis is 102% of 2568 MWt. (Also, the RCPPM setpoints were used in this analysis.)

Expected Doppler and Moderator Temperature Coefficient values are -1.5×10^{-5} and $-0.65\times10^{-4}\Delta k/k°F$, respectively, while values used in analysis are -1.27×10^{-5} and $0.0\Delta k/k°F$, respectively.

Expected value of $F_{\Delta H}$ is 1.44, while value used in analysis is 1.71.

The minimum DNBR obtained during this transient is 2.10 which is well above the 1.45 FSAR value and the 1.30 criterion. Fuel and cladding temperatures did not increase over the FSAR values. Without the power upgrade and without the installation of the RCPPM, the main difference between this analysis and the analysis submitted for the modified Cycle 1 (MC1) is the increased assumed core flow from 105% of design flow for MC1 to 106.5% of design flow for Cycle 2. This difference increases the DNBR margin over Cycle 1 analysis which is, therefore, bounding to Cycle 2.

3.3 Locked-Rotor

The locked-rotor accident is analyzed using the same conservative assumptions used in the four-pump coastdown transient discussed above.

The licensee concluded that less than 0.5% of the fuel pins in the core will experience a DNBR less than 1.30, and no pins will experience a DNBR less than 1.00. Also, the licensee concluded that for those pins that experience DNB, the cladding temperature will not exceed 1120°F. It is concluded that with less than 0.5% of the fuel pins experiencing a DNBR less than 1.30, Part 100 dose limits will not be reached by a wide margin and therefore it is acceptable.

3.4 Loss of Coolant Analysis

The loss of coolant accident (LOCA) has been previously analyzed at 2772 MWt (109% of 2544 MWt) and found acceptable in support of licensing and Cycle 1 operations. This analysis continues to be bounding for Cycle 2 operations at 2452 MWt and the intended increase to 2544 MWt.

As a result of a B&W small break analysis error identified in 1978, FPC has proposed, but not yet implemented, a permanent modification to eliminate the need for prompt local operator action in the event of a LOCA. This modification was approved on May 29, 1979. Since this modification has not been implemented, FPC proposed an exemption from 10 CFR 50.46 which is being addressed as a separate NRC staff action.

In addition, the Commission Order to FPC issued May 16, 1979, requires additional review of small breaks in the primary pressure boundary. FPC's response to the Order's requirements is also being addressed separately.

4.0 Startup Test Program

The startup test program proposed in Reference 1 was reviewed. 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, and (d) ejected control rod reactivity worth measurements. The power escalation test 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. We requested further information as to review criteria and remedial actions. The licensee responded in Reference 13. A symmetry test involving swapping of symmetrical rods was added to the zero-power test. Review criteria for the critical boron concentration measurement, the symmetry test, and the power distribution verification at 100% power were discussed and are stated in Reference 14.

We have reviewed the complete physics startup test program including review and acceptance criteria and remedial actions and find this program acceptable.

5.0 Evaluation of Technical Specification Changes

Proposed modifications to the CR-3 Technical Specifications are described below (References 1 and 12):

(1) Pressure/temperature limits would be changed due to higher assumed flow (106.5% of the design flow rate), use of BAW-2 CHF correlation, and as a result of IE Bulletin 79-05B (High Pressure Trip Setpoint).

(2) The flow rate would be increased to 106.5% of the design flow rate for consistency with other B&W reactors.

(3) Flux/Δ flux envelopes would be changed to allow higher power operation. However, higher power operation will be considered at a later date.

(4) Specifications 3.1.3.6, 3.1.3.7, 3.1.3.9, and 3.2.1 would reflect revised nuclear parameters as a result of the Cycle 2 reload.

(5) Table 2.2-1 would reflect the increased assumed flow.

(6) Table 3.2-1 would reflect the increased assumed flow.

(7) Table 3.2-2 would show error-adjusted limits to reflect the age of detectors.

(8) Regulating rod group insertion limits for 3 and 4 pump operation, and the axial power shaping and position limits would be provided for operation less than 233± 10 EFPD and for operation more than 233±10 EFPD. The 233 EFPD is the latest time in Cycle 2 at which the transient bank is nearly full-in at power level of 2452 MWt.

6.0 Conclusions

We have evaluated the reloading of CR-3 for Cycle 2 operation and the proposed Technical Specification modifications that reflect the new cycle parameters. In the original submittal, the licensee had intended to start Cycle 2 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, due to a licensee change of plans Cycle 2 will start at the same Cycle 1 power level of 2452 MWt.

After evaluating the FPC submittals (References 1, 2, 12), we conclude that CR-3 operation at or below 2452 MWt is acceptable.

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

Dated: July 3, 1979

References

1. Letter, W. P. Stewart, FPC to Director, ONRR, USNRC dated 2/28/79.
2. Letter, W. P. Stewart, FPC to Director, ONRR, USNRC dated 3/15/79.
3. Letter, W. P. Stewart, FPC to Director, ONRR, USNRC dated 11/24/78.
4. Clad Creep Collapse, BAW-10084P-A, Rev. 2, January 1979.
5. CR-3 Final Densification Report, BAW-1397, August 1973.
6. Fuel Pin Temperature & Gas Pressure Analysis, BAW-10044, May 1972.
7. Fuel Densification Report, BAW-10055, Rev. 1, July 1973.
8. Letter, K. E. Shurke, B&W to S. A. Varga, USNRC, "Densification Power Spikes," December 6, 1976.
9. Fuel Densification Report, Update of BAW-10055, December 5, 1977.
10. PDQ07 User's Manual, BAW-10-10117P-A, January 1977.
11. Letter, J. H. Taylor, B&W to D. B. Vassallo, USNRC, "Determination of the Fuel Rod Bow DNB Penalty," December 13, 1978.
12. Letter, W. P. Stewart, FPC to Director, ONRR, USNRC dated May 25, 1979.
13. Letter, FPC to USNRC dated 5/31/79.
14. Letter, FPC to USNRC dated 6/8/79.

TABLE 1 Applicability of FSAR Analysis to Cycle 2

Accident	Reference	Power Level MWt	Status of Analysis Relative to Cycle 2 Operation: Percent of 2452 MWt	Remark	Percent of 2544 MWt	Remark
Rod Withdrawal	FSAR	100% of 2568	105%	Bounding	101%	see footnote 1
Moderator Dilution	FSAR	100% of 2568	105%	Bounding	101%	see footnote 2
Cold Water (2-pump start)	FSAR	50% of 2568	52.5%	Bounding	50.5%	see footnote 3
4-PCD	Reload Report	102% of 2568	107%	Bounding	103%	Bounding
Locked Rotor	Reload Report	102% of 2568	107%	Bounding	103%	Bounding
Stuck in, Stuck out, Rod Drop	FSAR	100% of 2568	105%	Bounding	101%	see footnote 4
Loss of Electrical Power	FSAR	100% of 2568	105%	Bounding	101%	see footnote 5
SLB	FSAR	100% of 2568	105%	Bounding	101%	see footnote 1
S.G. Tube Rupture	FSAR	100% of 2568	105%	Bounding	101%	see footnote 5
Fuel Handling	FSAR	100% of 2568	105%	Bounding	101%	see footnote 5
Rod Ejection	FSAR	100% of 2568	105%	Bounding	101%	see footnote 1
Max. Hypothetical Accident	FSAR	100% of 2568	105%	Bounding	101%	see footnote 5
Waste Gas Tank Rupture	FSAR	100% of 2568	105%	Bounding	101%	see footnote 5
LOCA	Ref. 15,18,20	100% of 2772	113%	Bounding	109%	Bounding
Let Down Line Rupture Outside Containment	Reload Report	100% of 2603	106%	Bounding	102%	Bounding
MFWLB	FSAR	100% of 2568	105%	Bounding	101%	see footnote 2

Footnotes

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 MWt. The effect of a higher initial power of 102% of 2544 MWt (2595 MWt) is to cause the pressure trip to occur slightly sooner.

3. If the two pumps are started from a 51% full power the transient will produce a slightly higher pressure, thermal and neutron power increase. But since the FSAR analysis (run at 50.5% full power) produced maximum neutron power of 79%, maximum thermal power of 65% and only 150 psi increase over steady state pressure and other analysis assumptions, e.g., MTC and Doppler coefficient are conservative, the FSAR analysis is considered bounding.

4. Starting the transient at 102% of 2544 MWt would yield a slightly higher system pressure during the transient.

5. Starting the transient at 102% of 2544 MWt would yield approximately 1% higher doses than the FSAR values. However, that will still be much less than 10 CFR 100 limits.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of	)	
	)	
Florida Power Corporation, et al	)	Docket No. 50-302
	)	
Crystal River Unit No. 3, Nuclear Generating Plant	)	
	)	

EXEMPTION

I.

Florida Power Corporation (licensee) and eleven other co-owners are the holders of Facility Operating License No. DPR-72 which authorizes the operation of the nuclear power reactor known as Crystal River Unit No. 3 Nuclear Generating Plant (facility), at steady state reactor power levels not in excess of 2452 megawatts thermal (rated power). The facility consists of a Babcock & Wilcox (B&W) designed pressurized water reactor (PWR) located at the licensee's site in Citrus County, Florida.

II.

In April 1978 it was determined that the limiting small break loss of coolant accident (LOCA) for B&W 177 fuel assembly lowered loop plants was a break on the discharge side of a reactor coolant pump. For this break, assuming loss of off site power and the worst single failure, about 50% of the flow from one high pressure injection pump would be available to cool the core. Calculations indicated that peak clad temperature might exceed the 2200 F limit.

CR-3 was shutdown when this problem was identified. By letter dated June 14, 1978 the licensee submitted justification for restart and operation of the CR-3 at rated power prior to implementation of a permanent solution to the small break analysis problem.

This justification included procedure changes to direct operator actions in the event of a LOCA. These actions included prompt local operation of valves after the onset of a LOCA. The licensee also proposed a permanent solution for the small break analysis problem by letter dated July 21, 1978, as supplemented on July 27, 1978. In addition the licensee, by letter dated August 4, 1978, requested an exemption from the provisions of 10 CFR 50.46. The licensee stated that CR-3 could be operated to 100% rated power in full compliance with 10 CFR 50.46; however, to assure this, prompt local operator actions as described in their June 14, 1978 letter were necessary.

On September 1, 1978, Florida Power Corporation was granted an exemption from the provisions of 10 CFR Part 50, Paragraph 50.46(a). This exemption allowed continued operation of CR-3 in accordance with procedures described in the licensee's letter of June 14, 1978. The exemption was conditioned to terminate upon completion of the permanent solution to the small break analysis problem or completion of the remainder of operating Cycle 1, whichever occurred first.

The permanent solution proposed for CR-3 consists of modifications to provide a means of supplying electrical power to the motor operators of the high pressure injection valves from the Engineering Safeguards electrical busses of both channels. The licensee estimated an interval of 31 weeks between final NRC approval and installation of these modifications due to equipment procurement schedules. These proposed modifications were approved on May 29, 1979.

In anticipation of not being able to install an NRC approved modification prior to restart from the Cycle 2 reload, the licensee, by letter dated March 14, 1979 requested an exemption for Cycle 2 similar to that under which it was operating at the time. In response to an NRC staff request for an economic impact assessment of requiring that the modification be installed prior to restart from the current outage, the licensee provided, by letter dated June 5, 1979, the following:

1. The installation of the permanent modification is of the highest priority to the licensee and all avenues of expediting this installation are being employed including premium payment for rush orders.

2. Extending the current outage until installation of the modification is complete would delay restart until at least July 20, 1979 and would cost consumers about 15.3 million dollars in replacement fuel costs. This would also increase the probability of the licensee not being able to supply its firm load customers in June and July 1979.

3. Shut down of the facility upon receipt of the final piece of equipment would involve a two week outage and replacement fuel costs of about 6.3 million dollars. The final piece of equipment is scheduled to be received by July 6, 1979.

4. Facility operation until August 6, 1979 would permit installation of some equipment during power operation as well as allow for some slip in equipment delivery. Complete installation of the modification could be done in a 3 to 4 day outage, which could be scheduled for a weekend, and would involve about 1.8 million dollars in replacement fuel costs.

Based on the above, the licensee requested that the exemption from 10 CFR 50.46 be granted and that the facility be authorized to operate at rated core thermal power until August 6, 1979. In addition the licensee stated that the justification for restart and operation contained in their June 14, 1978 letter remains valid.

We have reviewed the effects of changes made to the facility during the current outage and have concluded that operation of CR-3 at power levels of up to 2452 Mwt and in accordance with the operating procedures of this Exemption, will assure that the ECCS system will conform to the performance criteria of 10 CFR 50.46. Accordingly, until modifications are completed to achieve full compliance with 10 CFR 50.46, operation of the facility at power levels up to 2452 Mwt with appropriate operating procedures will not endanger life or property or the common defense and security.

We have also reviewed the licensee's economic impact assessment and agree that economic factors justify the proposed installation schedule.

In the absence of any safety problem associated with the facility during the period until the modifications for achieving full compliance with 10 CFR 50.46 are completed, there appears to be no public interest consideration favoring undue restriction of the operation of the captioned facility. Accordingly, the Commission has determined that an exemption in accordance with 10 CFR 50.12 is appropriate. The specific exemption is limited to the period of time necessary to complete modifications regarding the ECCS system, but no later than August 6, 1979.

III.

Copies of the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, Washington, D.C. 20555, and are being placed in the Commission's local public document room at the Crystal River Public Library, Crystal River, Florida.

(1) B&W Report "Analysis of Small Breaks in the Reactor Coolant Pump Discharge Piping for the B&W Lowered Loop 177 FA Plants" dated May 1, 1978.

(2) The licensee's justification for restart and interim operation dated June 14, 1978.

(3) The application for exemption dated March 14, 1979.

(4) The licensee's clarification of exemption request dated June 5, 1979, and

(5) This Exemption in the matter of Florida Power Corporation, et al, Crystal River Unit No. 3 Nuclear Generating Plant.

IV.

WHEREFORE, in accordance with the Commission's regulations as set forth in 10 CFR 50.12, Florida Power Corporation is hereby granted an exemption from the provisions of 10 CFR Part 50, Paragraph 50.46(a). With respect to Crystal River Unit No. 3, this exemption is conditioned as follows:

(1) Until further authorization by the Commission, Florida Power Corporation shall operate in accordance with the procedures described in its letter of June 14, 1978.

(2) This exemption shall be terminated upon completion of the modifications in accordance with this exemption or August 6, 1979, whichever occurs first.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut, Acting Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland,
this 3rd day of July 1979.

7590-01

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. 19 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.

This amendment revises the Technical Specifications to permit power operation during Cycle 2 at the currently authorized power level of 2452 MWt.

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The applications for the amendment comply 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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with Cycle 2 operation at an increased power level of 2544 MWt was published in the FEDERAL REGISTER on March 28, 1979 (44 FR 18569). No request for a hearing or petition for leave to intervene was filed following this notice of proposed action. At the licensee's request, the Commission has postponed action on the power increase.

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 applications for amendment dated February 28, 1979 and March 15, 1979, as supplemented May 25, 1979, (2) Amendment No. 19 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 Operating Reactors.

Dated at Bethesda, Maryland, this 3rd day of July 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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