September 1, 1978

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Docket No. 50-302

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

Gentlemen:

The Commission has issued the enclosed Amendment No. 16 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 response to your applications dated June 6, and July 21, 1978.

This amendment revises the Technical Specifications to support operation of Crystal River Unit No. 3 during the remainder of Cycle 1 with 4 fuel assemblies acquired from Oconee Unit 1 and without Burnable Poison Rod Assemblies and most Orifice Rod Assemblies. In addition, Quadrant Power Tilt limits are changed to compensate for increased detector uncertainties and administrative use of revised reactor coolant pressure-temperature limits has been reviewed.

As requested in your letter dated August 4, 1978, the Commission has also issued the enclosed Exemption for Crystal River Unit No. 3 from the requirements of 10 CFR 50.46(a)(1).

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. A copy of the Exemption is being filed with the Office of the Federal Register for publication.

Sincerely,

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

Enclosures and cc:
See next page

Const. 1
GD

RSB:DOR	EB:DOR	OELD
PCheck*	VNoonan*	SLewis*
9/ /78	9/ /78	9/ /78

*SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE➤	ORB#4:DOR	ORB#4:DOR	C-ORB#4:DOR	AD-E&P:DOR	D:DOR	NRR
SURNAME➤	RIngram*	CNelson:dn	RWReid	BGrimes	VStello	EGCase
DATE➤	9/ /78	9/1/78	9/1/78	9/1/78	9/1/78	9/1/78

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As requested in your letter dated August   , 1978, the Commission has also issued the enclosed Exemption for Crystal River Unit No. 3 from the requirements of 10 CFR 50.46(a)(1) that Emergency Core Cooling System performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K to 10 CFR 50.

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed. A copy of the Exemption is being filed with the Office of the Federal Register for publication.

Sincerely,

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

RS:DOR PCHECK 9/1/78	EB:DOR VNoonan 9/1/78	AD-E&P:DOR BGrimes 9/ /78	D:DPR VStello 9/ /78

~~Enclosures and cc:~~
See next page

Concurred subject to changes as discussed with Chris Nelson.

OFFICE	ORB#4:DOR	ORB#4:DOR	OELD	C-ORB#4:DOR	NRR
SURNAME	RIngram	CNelson:dn	SHLewis	RWReid	EGCase
DATE	9/1/78	9/1/78	9/1/78	9/ /78	9/ /78

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. 16 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 of 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 support operation of Crystal River Unit No. 3 during the remainder of Cycle 1 with 4 fuel assemblies acquired from Oconee Unit 1 and without Burnable Poison Rod Assemblies and most Orifice Rod Assemblies. In addition, Quadrant Power

Tilt limits are changed to compensate for increased detector uncertainties and administrative use of revised reactor coolant pressure-temperature limits has been reviewed.

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. 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 applications for amendment dated June 6, and July 21, 1978, (2) Amendment No. 16 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 1st day of September 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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

Enclosures:
1. Amendment No. 16
2. Exemption
3. Safety Evaluation
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

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

Chief, Energy Systems Analyses
Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S.W.
Washington, D. C. 20460

Crystal River Public Library
Crystal River, Florida 32629

cc w/enclosures and incoming
dtd: 6/6, 7/21/ & 8/4/78
Bureau of Intergovernmental Relations
660 Apalchee Parkway
Tallahassee, Florida 32304

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
Montgomery Building
2562 Executive Center Circle, E.
Tallahassee, Florida 32301

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

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
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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. 16
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 June 6, and July 21, 1978, 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.

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

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 16 , 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 Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: September 1, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 16

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

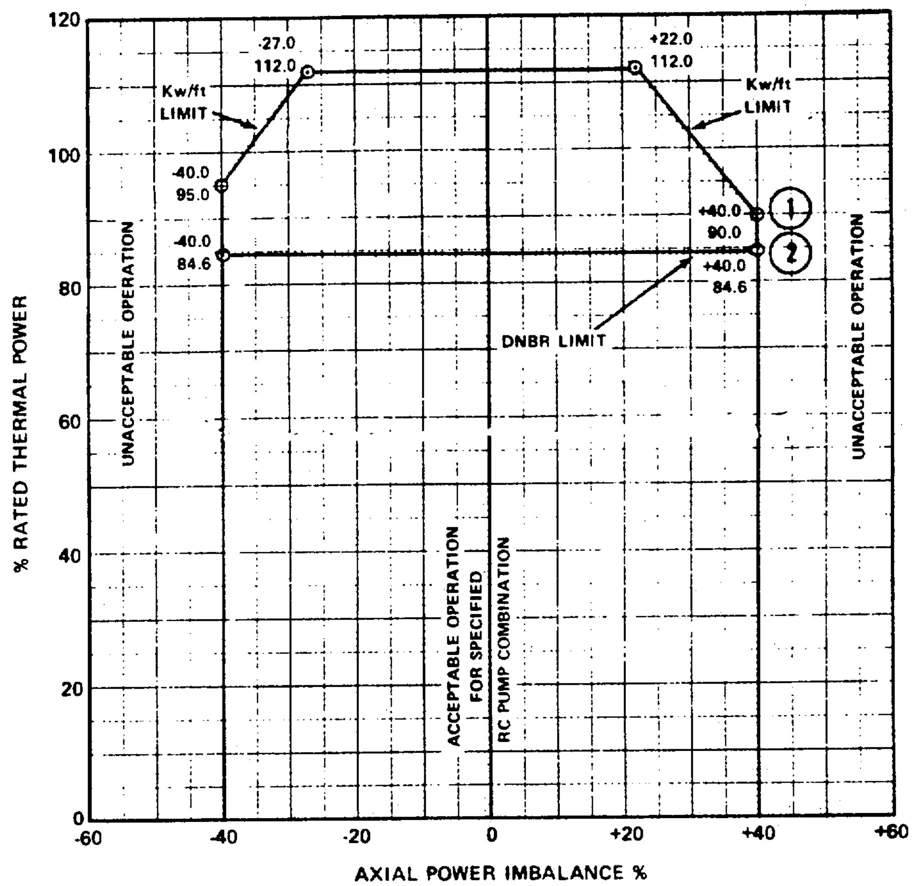
Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Pages

III
2-3
B 2-1 thru B 2-7
3/4 1-14
3/4 1-16
3/4 1-27 thru 3/4 1-32
3/4 1-37 (added)
3/4 1-38 (added)
3/4 2-2
3/4 2-3
3/4 2-6
3/4 2-11
B 3/4 1-2
B 3/4 1-4
B 3/4 2-1
B 3/4 2-2
5-4

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS



CURVE	REACTOR COOLANT FLOW (1b/hr)
1	137.9×10^6
2	103.0×10^6

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.

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 137.89×10^6 lbs/hr, which is 105% 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, three pumps, and one pump in each loop, 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.

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

2.1.3 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

2.2.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

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

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

Manual Reactor Trip

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

Nuclear Overpower

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

During normal station operation, reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which was used in the safety analysis.

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%.

3. Trip would occur when one reactor coolant pump is operating in each loop (total of two pumps operating) if the power is $\geq$ 51.3% and reactor flow rate is 49.2% or flow rate is $\leq$ 47.9% and the power level is 50.0%.

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 1.043% 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.

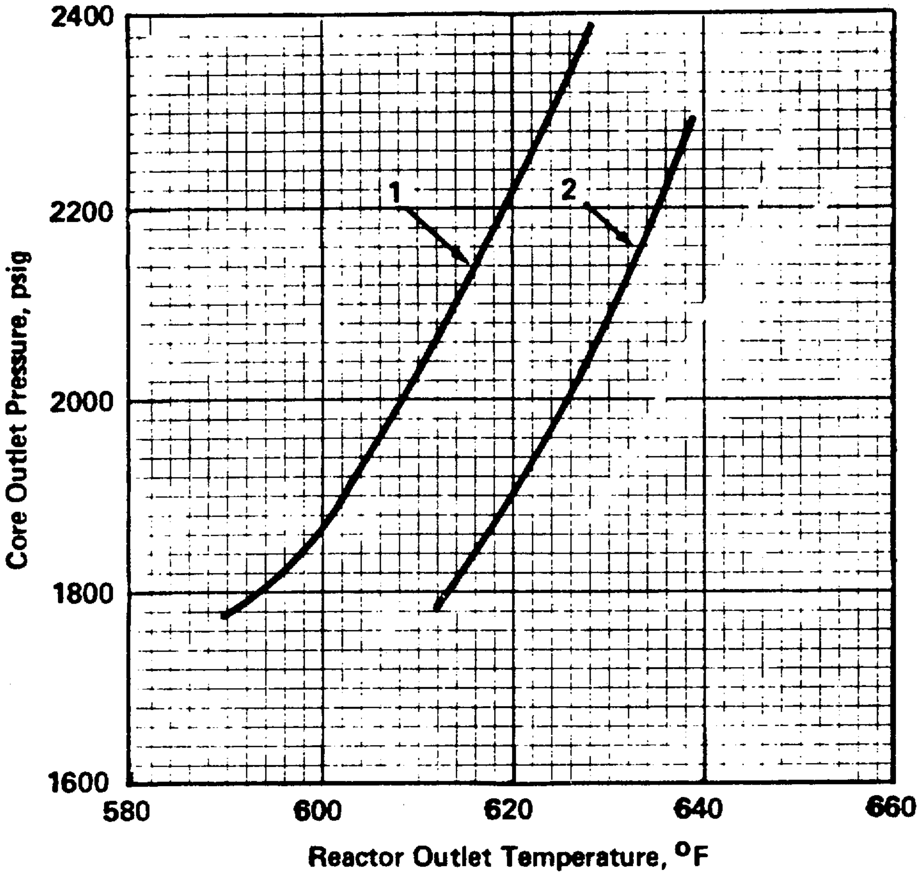
During a slow reactivity insertion startup accident from low power or a slow reactivity insertion from high power, the RCS Pressure-High setpoint is reached before the Nuclear Overpower Trip Setpoint. The trip setpoint for RCS Pressure-High, 2355 psig, has been established to maintain the system pressure 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, ($16.25\ T_{out}$°F-7838) 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 ($16.25\ T_{out}$°F-7878) psig.

Reactor Containment Vessel Pressure - High

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



Reactor Coolant Flow

Curve	(LBS/HR)	Power	Pumps Operating (Type of Limit)
1	137.9×10^6 (105%)	112%	Four Pumps (DNBR Limit)
2	103.0×10^6 (74.7%)	85.9%	Three Pumps (DNBR Limit)

Pressure/Temperature Limits at Maximum
Allowable Power for Minimum DNBR

BASES Figure 2.1

BORIC ACID PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.7 At least one boric acid pump in the boron injection flow path required by Specification 3.1.2.2a shall be OPERABLE and capable of being powered from an OPERABLE emergency bus if the flow path through the boric acid pump in Specification 3.1.2.2a is OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With no boric acid pump OPERABLE, restore at least one boric acid pump to OPERABLE status within 72 hours or be in at least HOT STANDBY and borated to a SHUTDOWN MARGIN equivalent to 1% $\Delta k/k$ at 200°F within the next 6 hours; restore at least one boric acid pump to OPERABLE status 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 5500 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 5500 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:

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.

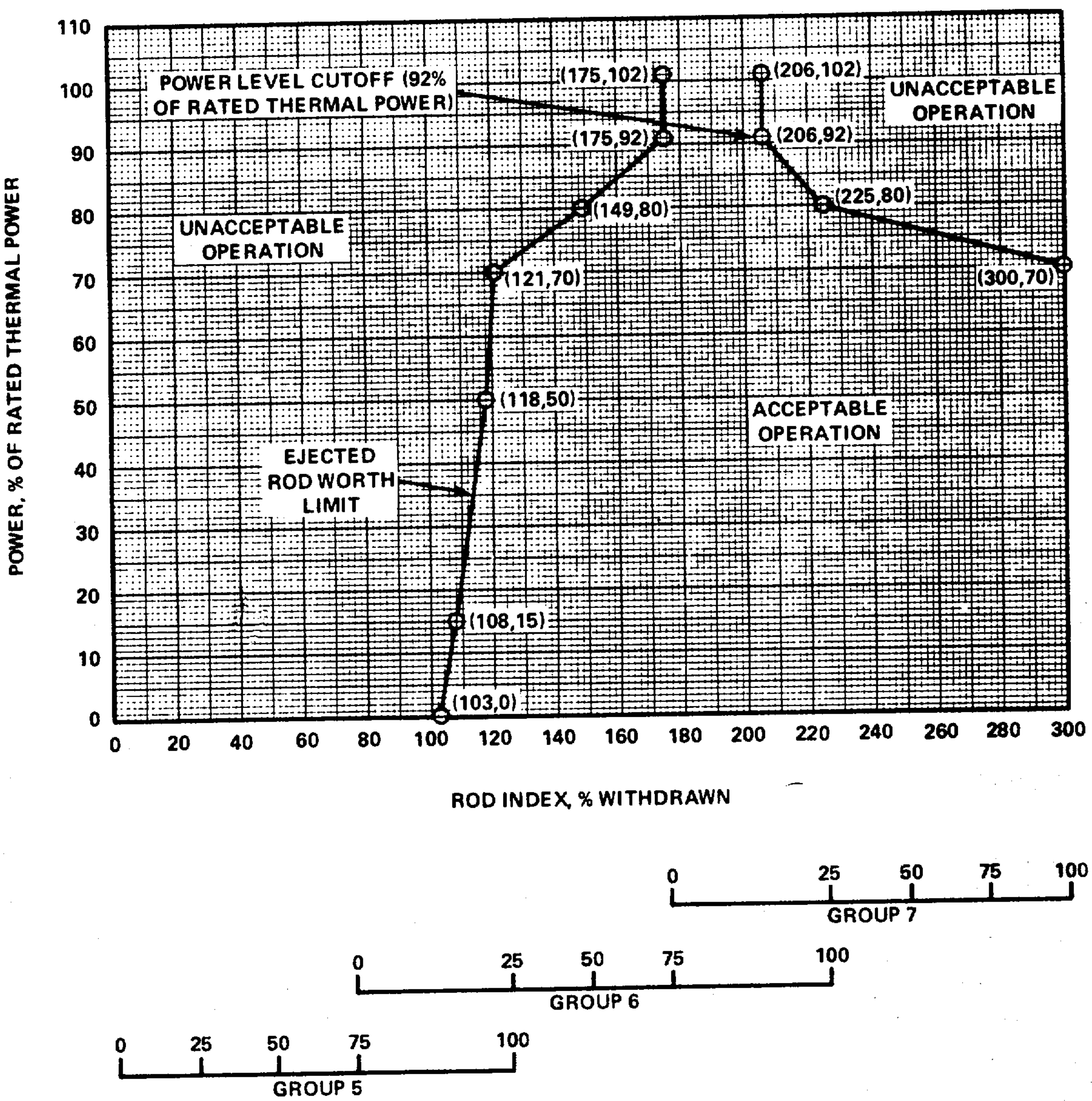


Figure 3.1-1

Regulating Rod Group Insertion Limits for 4 Pump Operation From 268.8 EFPD to 400 ± 10 EFPD

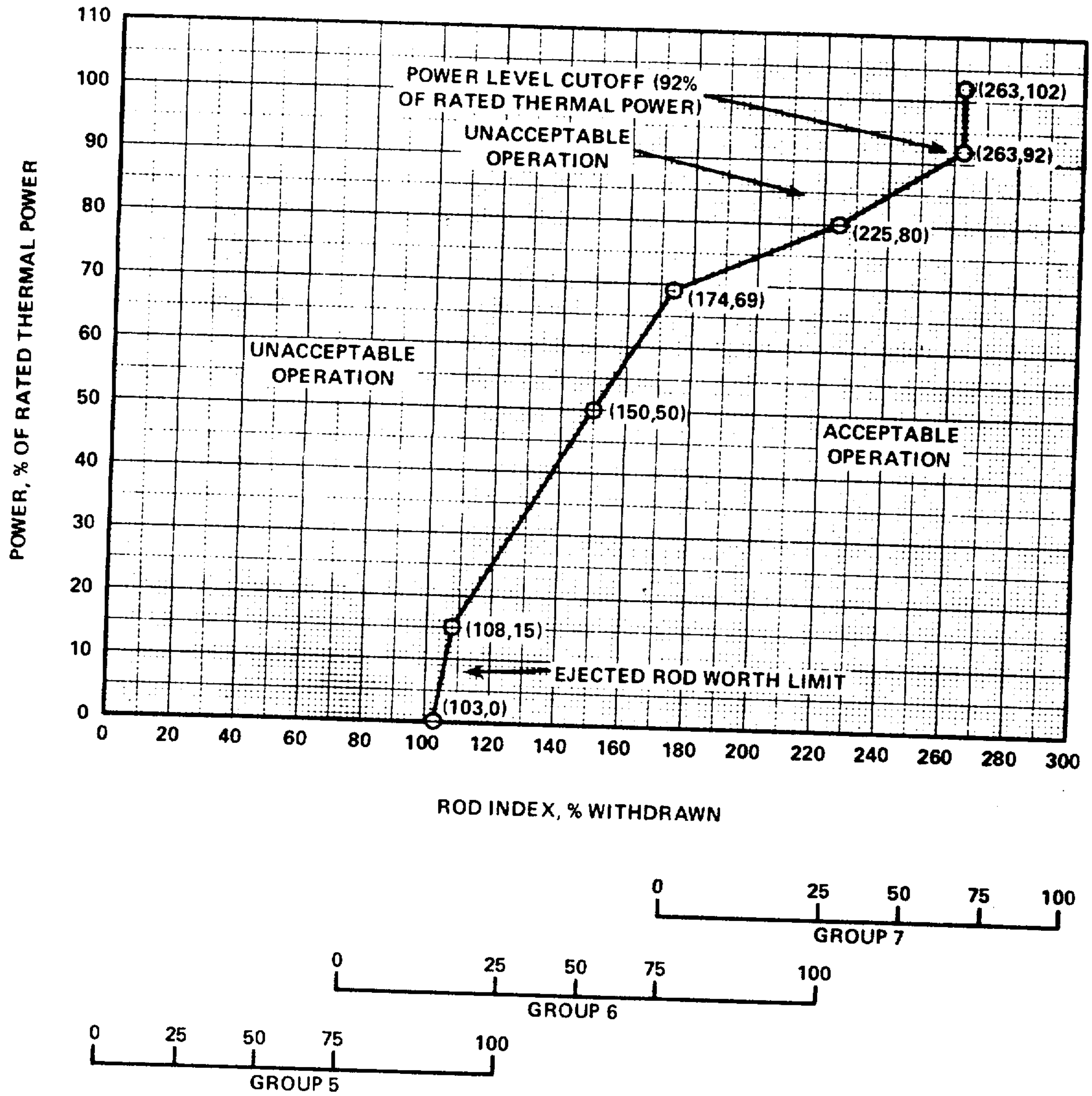


Figure 3.1-2

Regulating Rod Group Insertion Limits for 4 Pump Operation After 400 ± 10 EFPD

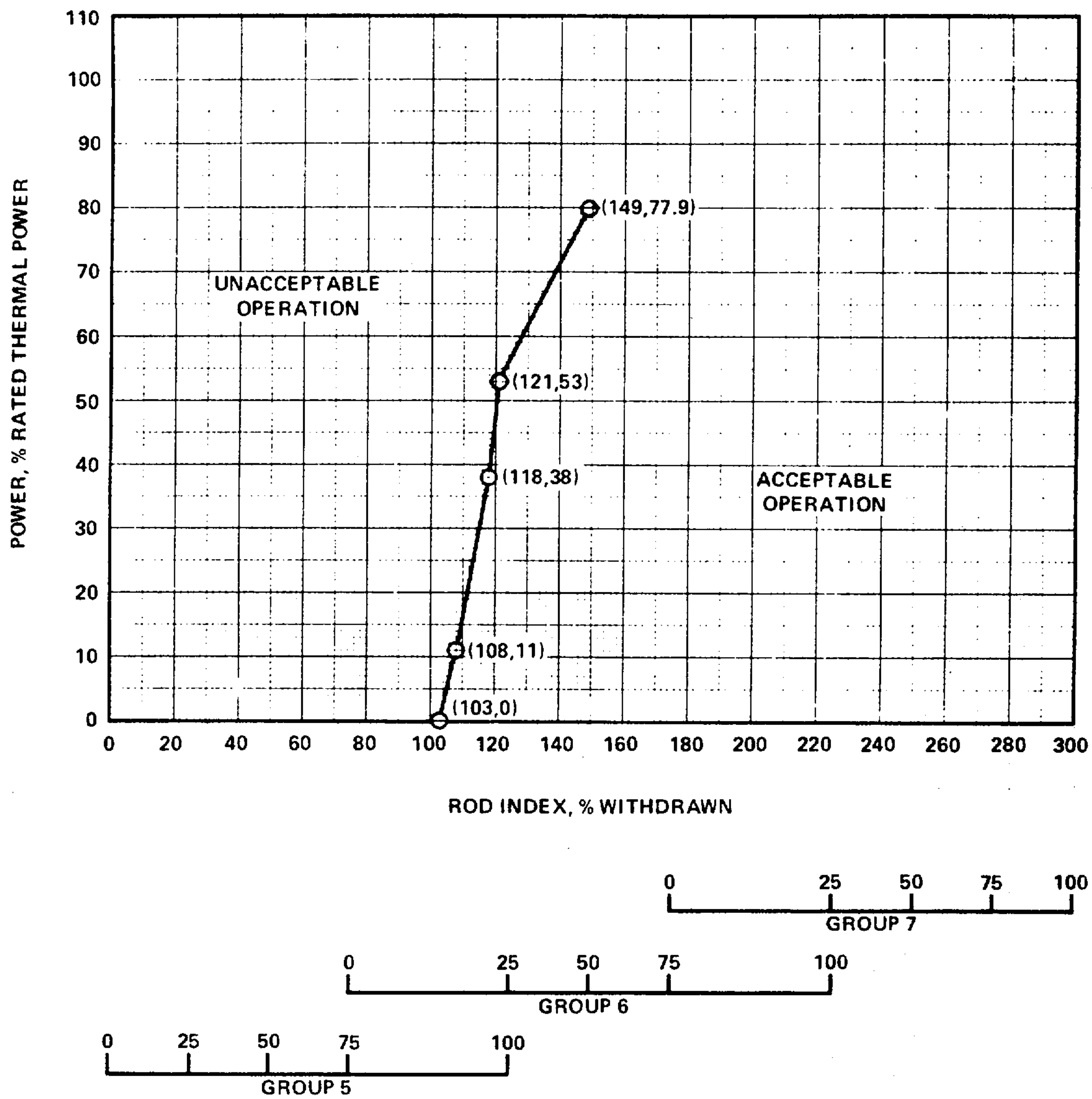


Figure 3.1-3

Regulating Rod Group Insertion Limits for 3 Pump Operation from 268.8 to 400 ± 10 EFPD

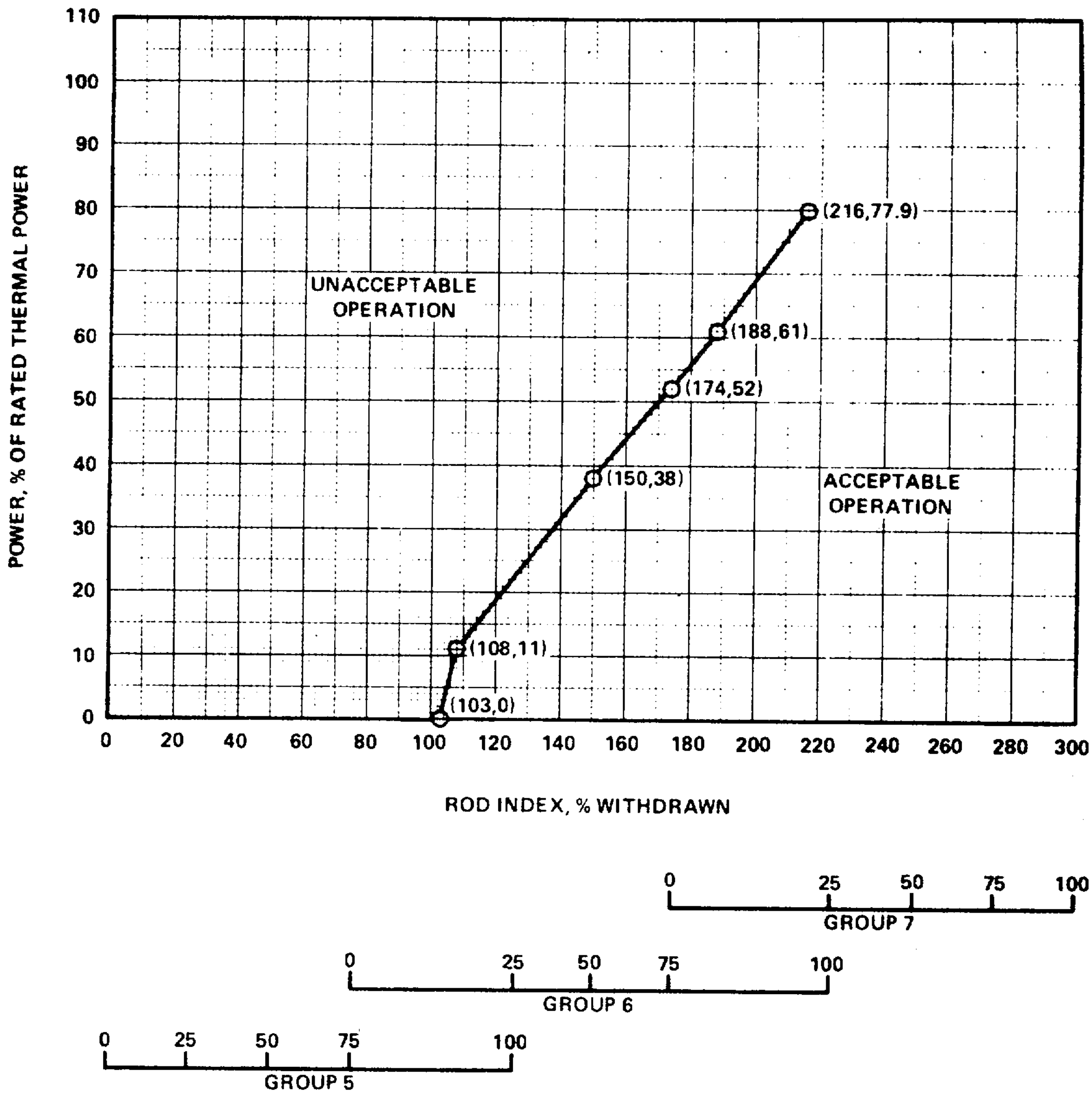


Figure 3.1-4

Regulating Rod Group Insertion Limits for 3 Pump Operation After 400 ± 10 EFPD

DELETED

DELETED

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 Figure 3.1-9.

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 $k_{eff} \geq 1.0$.

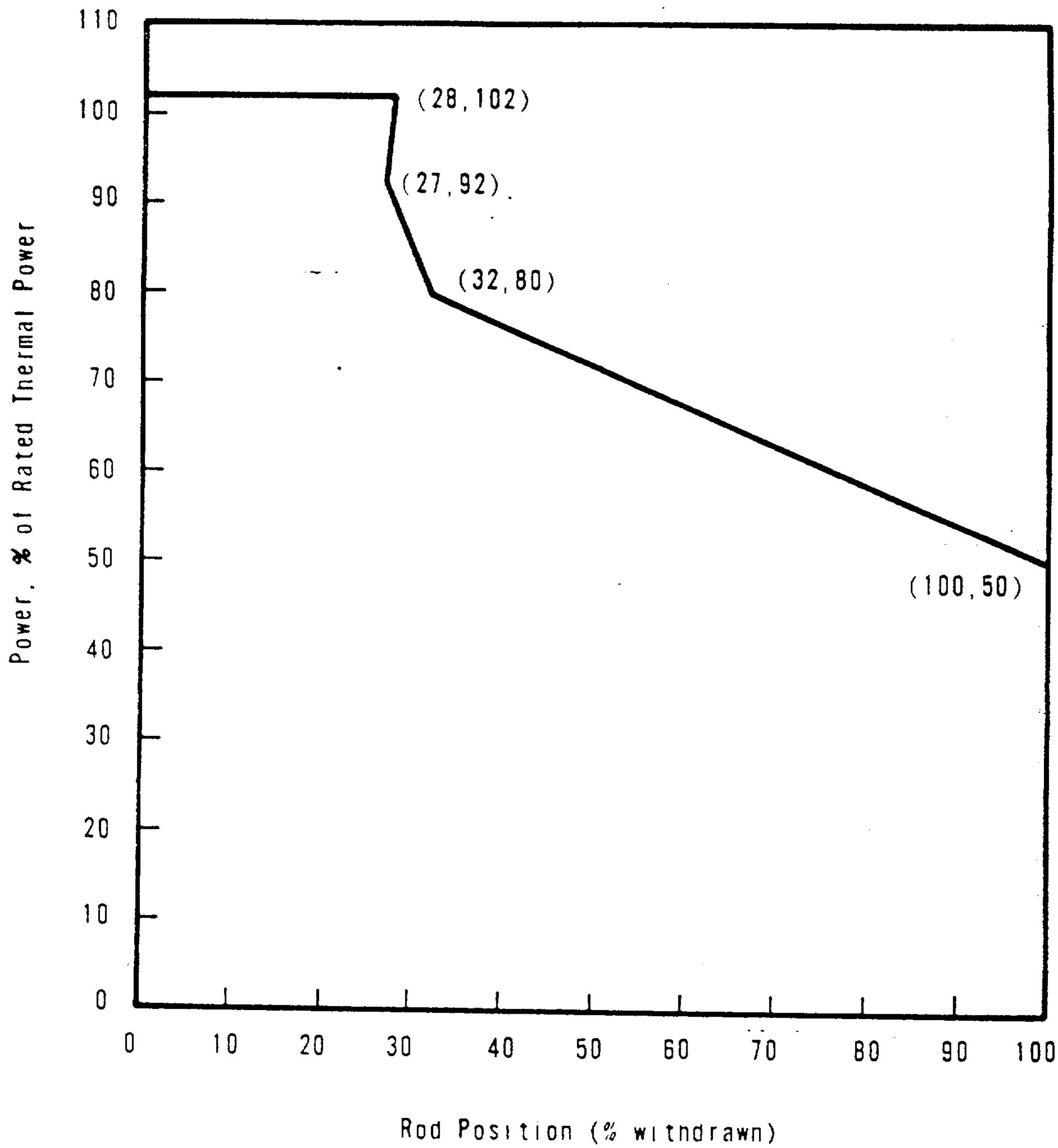


Figure 3.1-9

Axial Power Shaping Rod Group Insertion Limits
After 268.8 EFPD

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL POWER IMBALANCE

LIMITING CONDITION FOR OPERATION

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

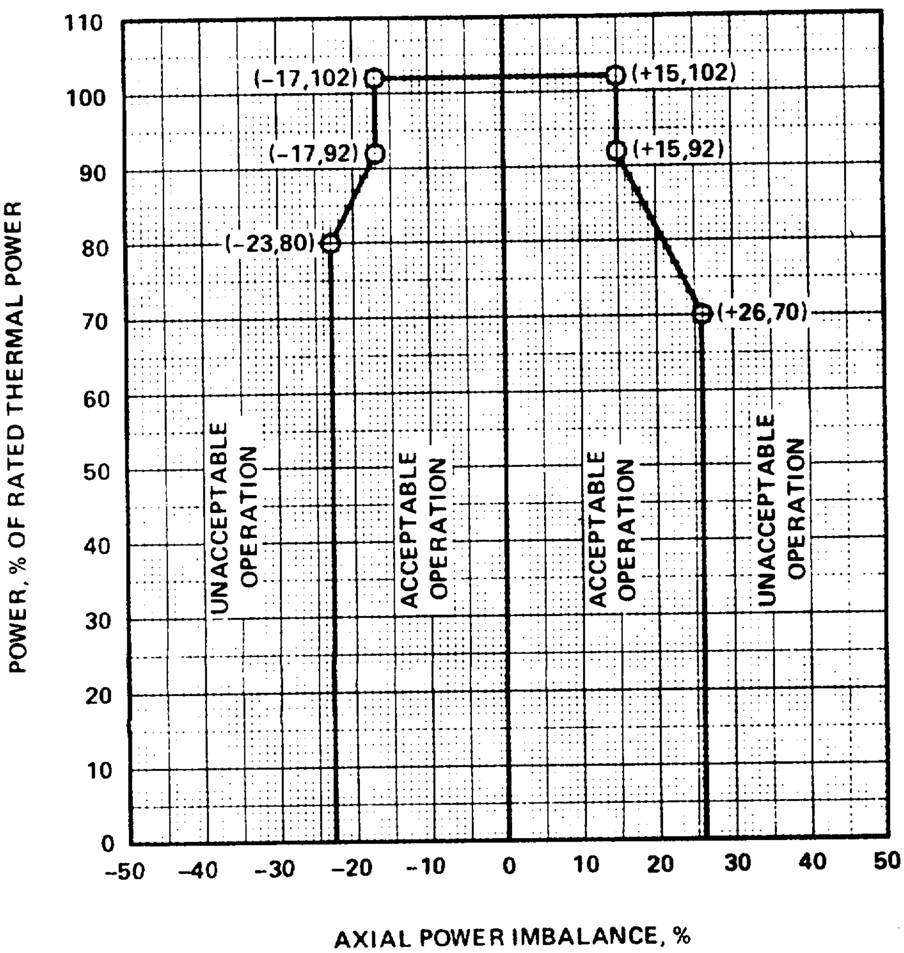


Figure 3.2-1

Axial Power Imbalance Envelope for Operation
After 268.8 EFPD

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NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - F_Q

LIMITING CONDITION FOR OPERATION

3.2.2 F_Q shall be limited by the following relationships:

$$F_Q \leq \frac{3.12}{P}$$

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

APPLICABILITY: MODE 1.

ACTION:

With F_Q exceeding its limit:

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

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

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

SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE REQUIREMENTS (Continued)

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

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

c. The provisions of Specification 4.0.4 are not applicable.

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

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

LIMITING CONDITION FOR OPERATION

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

$$F^N_{\Delta H} \leq 1.71\ [1 + 0.6(1-P)]$$

where $P = \dfrac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$

and $P \leq 1.0$

APPLICABILITY: MODE 1.

ACTION:

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

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

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

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

TABLE 3.2-2

QUADRANT POWER TILT LIMITS

	STEADY STATE LIMIT	TRANSIENT LIMIT	MAXIMUM LIMIT
Measurement Independent QUADRANT POWER TILT	4.92	11.07	20.0
QUADRANT POWER TILT as Measured by:			
Symmetrical Incore Detector System	3.61	9.11	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.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 BORATION CONTROL

3/4.1.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. During Modes 1 and 2 the SHUTDOWN MARGIN is known to be within limits if all control rods are OPERABLE and withdrawn to or beyond the insertion limits.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident a minimum SHUTDOWN MARGIN of 0.60% $\Delta k/k$ is initially required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required is based upon this limiting condition and is consistent with FSAR safety analysis assumptions.

3/4.1.1.2 BORON DILUTION

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

3/4.1.1.3 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. The surveillance 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 1.0% $\Delta k/k$ 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 5210 gallons of 12,250 ppm borated water from the boric acid storage tanks or 32,536 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

3/4.1.2 BORATION SYSTEMS (Continued)

stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% of $\Delta k/k$ after xenon decay and cooldown from 200°F to 140°F. This condition requires either 165 gallons of 12,250 ppm borated water from the boric acid storage system or 888 gallons of 2270 ppm borated water from the borated water storage tank.

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.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and during short term transients, (b) maintaining the peak linear power density $\leq$ 18.0 kw/ft during normal operation, and (c) maintaining the peak power density $\leq$ 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 Figure 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:

$$F_Q \leq 3.12; \quad F^N_{\Delta H} \leq 1.71$$

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.

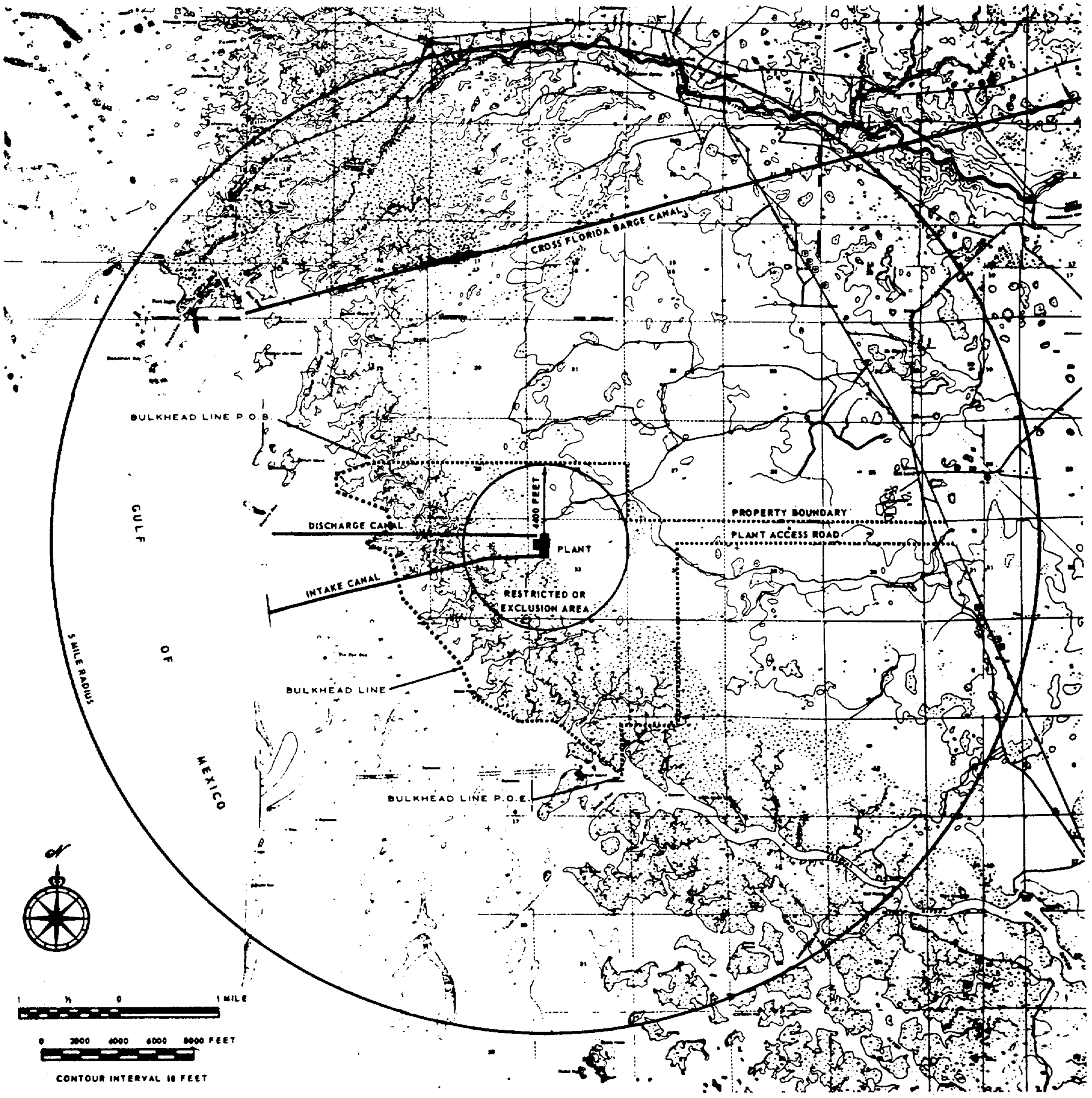
2. Regulating rod groups are sequenced with overlapping groups as required in Specification 3.1.3.6.

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 between +15 percent and - 17 percent at RATED THERMAL POWER.

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

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



LOW POPULATION ZONE

FIGURE 5.1-2

DESIGN PRESSURE AND TEMPERATURE

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

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2229 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.50 weight percent U-235.

CONTROL RODS

5.3.2 The reactor core shall contain 61 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber material. The APSR's shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

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 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 (CR-3), 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 licensees' site in Citrus County, Florida.

II.

In accordance with the requirements of the Commission's Emergency Core Cooling System (ECCS) Acceptance Criteria, 10 CFR 50.46, Florida Power Corporation (FPC) submitted on August 6, 1974, an ECCS evaluation for the facility. The ECCS performance submitted by FPC was based upon an ECCS Evaluation Model developed by B&W, the designer of the Nuclear Steam Supply System for this facility. The B&W ECCS Evaluation Model had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46, and Appendix K. The evaluation indicated that with the limits set forth in CR-3's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On April 12, 1978, B&W informed the NRC that it had determined that in the event of a small break Loss of Coolant Accident (LOCA) on the discharge side of a reactor coolant pump, high pressure injection (HPI) flow to the core could be reduced somewhat. Subsequent calculations indicated that in such a case the calculated peak clad temperature might exceed 2200°F.

Previous small break analyses for B&W 177 fuel assembly (FA) lowered loop plants had identified the limiting small break to be in the suction line of the reactor coolant pump. Recent analyses have shown that the discharge line break is more limiting than the suction line break. As a result, it was necessary that operating B&W plants of this design provide justification and propose restrictions as necessary to continue operating. Since CR-3 was shutdown at that time, and has been since March 3, 1978, to conduct repairs resulting from burnable poison rod assembly failures, no immediate action was necessary.

CR-3 has an ECCS configuration which consists of two HPI trains. Each train has a HPI pump and injects into two of the four reactor coolant system (RCS) cold legs on the discharge side of the RCS pump. (There is also a third HPI pump installed.) The two parallel HPI trains are connected but are kept isolated by a manual valve (known as a cross-over valve) that is normally closed. Upon receiving a safety injection signal, the HPI pumps are started and valves in the four injection lines are opened. Assuming loss of offsite power and the worst single failure, only one HPI pump and two of the four injection paths would be available.

If a small break is postulated to occur in the RCS piping between the RCS pump discharge and the reactor vessel, the HPI flow injected into this line (about 50% of the output of one HPI pump) could flow out the break. Therefore, for the worst combination of break location and single failure, only 50% of the flow rate of a single HPI pump would contribute to maintaining the coolant inventory in the reactor vessel. This situation had not been previously analyzed and B&W had indicated that the limits specified in 10 CFR 50.46 may be exceeded.

B&W has stated that they have analyzed a spectrum of small breaks in the pump discharge line and have determined that to meet the limits of 10 CFR 50.46(b), operator action is required to open the manually operated crossover valve and to manually align the motor driven isolation valves which had failed to open. This would allow the flow from the HPI pump to feed all four reactor coolant legs. B&W has assumed that 30% of the flow would be lost through the break and 70% would enter the core.

B&W has prepared and submitted a summary entitled "Analysis of Small Breaks in the Reactor Coolant Pump Discharge Piping for the B&W Lowered Loop 177 FA Plants," May 1, 1978 (the B&W Summary), which describes the methods used and the results obtained in the above analysis. The analysis models operator action by assuming a step increase in flow to the reactor vessel (with balanced flow in the three intact loops) ten minutes after the LOCA reactor protection system trip signal occurs.

The results of the B&W analyses for reactor coolant pump discharge line break sizes of 0.17, 0.15, 0.13, 0.1, 0.07, and 0.04 ft^2 at a reactor power level of 2568 Mwt were presented in the B&W Summary of May 1, 1978. This power level is representative of the full power rating of similar B&W-designed reactors and encompasses the 2452 Mwt full power rating of CR-3. Based on these results, B&W states that with operator action consistent with that modeled in the analysis, a 0.13 ft^2 discharge line break is the most limiting case. In this case, core uncovery occurs for about 350 seconds and the conservatively calculated peak clad temperature is approximately 1550^oF. This temperature is well below the limit specified in 10 CFR 50.46(b).

Based on our review of the B&W analyses, we found that the calculations supported the conclusion that a .13ft^2 discharge line break is the most limiting case. However, the analyses did not demonstrate that the assumptions employed in supplying heat inputs to the FOAM Code (a part of the approved B&W model) were conservative. Therefore, B&W was requested to justify the use of these assumptions. By letter dated August 11, 1978, B&W submitted a report which describes and compares the simplified and detailed input methods for the FOAM analysis. We have reviewed this report and have determined that the assumptions used in supplying heat inputs are conservative and the use of the simplified input in the FOAM calculations meets the requirement for calculations using an approved model.

By letter dated June 14, 1978, FPC submitted justification for restart and operation of CR-3 at rated power prior to implementation of a permanent solution to the ECCS small break analysis problem. This submittal references the B&W Summary of May 1, 1978, stating that the results are applicable to CR-3, and presents procedural modifications to describe how the required operator actions have been instituted. These procedural changes regarding operator action are consistent with the assumptions of the B&W analysis and have been implemented. The letter also states that shift review and simulator training will be conducted on the changes. FPC has stated that this action has been completed. Based on the above, we conclude that the procedures implemented and the training conducted by FPC relative to operator action in the event of a small break are acceptable and allow reliance on prompt operator action for an interim period. We do not, however, consider the need to assume prompt operator action outside the control room to be in compliance with 10 CFR 50.46.

Reliance on prompt local operation of valves after the onset of a LOCA is not desirable on a permanent basis. FPC has, by letter dated July 21, 1978, as supplemented on July 27, 1978, proposed a permanent solution for this issue. We are currently reviewing this proposal.

Due to our inability to conclude that operation of CR-3 to 100% of licensed power would be in conformance with 10 CFR 50.46, and due to the pending restart of the facility, FPC was requested by telephone on July 24, 1978, to either provide an acceptable ECCS for resumption of Cycle 1 operation or request an exemption from the provisions of 10 CFR 50.46 with justification to support licensing of CR-3 operation.

By letter dated August 4, 1978, FPC requested an exemption from the provisions of 10 CFR 50.46. FPC states that CR-3 can be operated to 100% rated power in full compliance with 10 CFR 50.46; however, to assure this, prompt operator actions as described in FPC's June 14, 1978, letter are necessary. FPC also states that improvements to the ECCS prior to restart of CR-3 would not be feasible and therefore the exemption has been requested.

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.

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 the next scheduled refueling outage, currently scheduled to begin on May 1, 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) FPC's justification for restart and interim operation dated June 14, 1978.

(3) The application for exemption dated August 4, 1978,

(4) B&W comparison of amplified and detailed input methods for FOAM analysis dated August 11, 1978, 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 upon shutdown for the next scheduled refueling outage, whichever occurs first.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Dated at Bethesda, Maryland
this 1st day of September 1978.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. TO LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

Introduction

By letter dated June 6, 1978,(1) Florida Power Corporation (the licensee) requested an amendment to the Technical Specifications for Crystal River Unit 1 (CR-3). The amendment would allow continued operation of CR-3 without the use of Burnable Poison Rod Assemblies (BPRA) and Orifice Rod Assemblies (ORA), and also with repair of the damage caused by two ejected BPRAs. The licensee supplemented this request by letter dated July 21, 1978(2), to allow operation with four assemblies from another reactor to replace an assembly damaged during fuel handling.

As a separate issue, the licensee's June 6, 1978, submittal also requested a change in the Technical Specifications to reduce the maximum allowable value of neutron flux tilt as measured in each quadrant of the reactor core by in-core or out-of-core detectors, and move the measurement independent values to the bases.

On August 4, 1978, we were informed by Babcock & Wilcox (B&W) that weld filler wire, which may have been used in the manufacture of some B&W reactor vessels, has nickel and silicon contents outside the specified ranges. Since this departure could affect the stress capabilities of a reactor vessel, applicable B&W facilities were requested to report their actions and intentions in this matter. Florida Power Corporation responded by letter dated August 18, 1978.

We have evaluated the licensee's Technical Specification changes proposed in support of the CR-3 restart and we have evaluated the interim measures taken by the licensee to address the atypical weld wire issue.

EVALUATION

1.0 BPRA Failures

On March 3, 1978, at 2337 hours, CR-3 was shut down to investigate an apparent loose part(s) detected by the Loose Parts Monitoring System (LPMS) in the upper tube sheet region of the "B" once-through steam generator (OTSG). Inspection(3) confirmed that several pieces of a BPRA were in the "B" OTSG and subsequent video inspection revealed damage in the tube and tube sheet area. Inspections of the Reactor Vessel Internals also revealed that an additional adjacent BPRA had become separated from its fuel assembly and was in the plenum. The reactor was completely defueled and placed in an extended outage to evaluate the causes and effects of the BPRA separations.

Following defueling, all 66 remaining BPRAs were subjected to a lock test, and all were found to be locked in their respective fuel assemblies. During removal of the 66 BPRAs, all ball-lock couplings were visually examined; nothing unusual was observed. Nine (9) of the BPRAs were visually examined full-length and 360° around on the inside. Two wear areas were observed on each latch assembly, oriented at 180° to each other.

Three fuel assemblies had wear in the holddown latches which approximated that observed in the holddown latches of the two fuel assemblies from which the BPRAs had separated.

The ORAs were also examined because they have the same holddown latch assembly mechanism as the BPRAs. None of the ORA holddown latch assemblies had wear marks, or any features except for two tiny spherical dimples corresponding to the location of the latching balls.

As a result of the evaluation of the anomalous mechanical behavior of the BPRAs, the licensee felt it prudent to remove all the BPRAs and all but two of the ORAs from CR-3 before the completion of the first cycle of operation.(4) The removal of the BPRAs and ORAs will result in a change in various nuclear parameters as well as resulting in an increase in core bypass flow. The two remaining ORAs will be modified ORAs (MORA) located in two fuel assemblies containing the primary neutron sources.

1.1 Cause of BPRA Failures

Results of the licensee's investigation of the CR-3 event indicate that the separation of BPRAs is due primarily to a long term wear phenomenon causing separation of the BPRA holddown latch. Coolant flow and the resultant net hydraulic lift compared with the wet weight of a BPRA is reported by the licensee(5) to be the primary factor in the holddown latch wear rate.

1.2 Corrective Action to Eliminate Future Failures

For the Cycle 1 restart, all unmodified ORAs and BPRAs will be removed from the core. This will leave 106 assemblies with open guide tubes and will increase the maximum total core coolant bypass flow rate from 6.04% to 10.40% of Reactor Coolant System (RCS) flow for the remainder of Cycle 1 operation. The removal of the BPRAs after 268.8 Effective Full Power Days (EFPD) of burnup has a relatively small impact on the core since the BPRAs have expended most of their neutron absorbing ability at this point in core life. Evaluation of the thermal-hydraulic effects is provided in Section 1.8.

As stated above the two modified ORAs will be located in the two fuel assemblies containing the primary neutron sources. The primary source capturing arrangement is modified to prevent the ORA from causing wear of the fuel assembly end fitting and coming loose. Twelve of the ORA rods are removed from the assembly, leaving only the rod above the source and the three symmetrically located rods. A canned spring pack device (retainer) is placed over the hub of the modified ORA and held down by the reactor internals. The spring is designed to hold the modified ORA firmly against the fuel assembly end fitting taking into account hydraulic lift, differential thermal expansion, and fuel assembly irradiation growth. Testing and analyses have shown hydraulic and structural adequacy of the retainer.

The design and testing of this retainer device are described in Reference 6. From a mechanical design standpoint, the basic concern is whether the retainer provides enough holddown force to preclude loosening of the MORAs. From analyses of the static and dynamic stresses on the retainer spring load arm and housing, results of prototype testing in a flow test facility, and in-air mechanical tests, criteria for use of the retainer with modified ORAs were established. The primary criterion is that the margin to component lift with the retainer, taking into account the hydraulic forces acting on the MORA, the MORA weight, and the retainer holddown force, should be greater than 30 pounds. This criterion is met with acceptable margin by the fact that when the retainer device is used with the modified ORA, the holddown force is greater than 35 pounds with all four reactor coolant pumps operating. A second criterion is related to fuel assembly irradiation growth. The fuel assembly burnup design value was used as a basis for the retainer design. Since the maximum burnup used in one cycle of operation will be less than the burnup used as a design basis, the fuel assembly growth criterion is met (note that the retainer will be used for only one cycle of operation).

The potential consequences of a retainer failure have also been addressed[6] although failure is considered unlikely. The neutronic and thermal-hydraulic consequences are considered insignificant. Interference with control rod motion, for example, would not, according to analyses of stuck-out control rod transients for B&W 177-FA plants, prevent safe shutdown of the plant.

The major concern associated with subsequent retainer failure is plant damage and potential outages for repair. This damage should be precluded by the LPMS since loose parts in general will be detected. Two sensors on the reactor vessel head and two on each OTSG upper tubesheet will be set to alarm for an impact energy of 0.5 ft/lb at a minimum distance of three feet. This setpoint is in agreement with NRC Regulatory Guide 1.133.

In summation, we conclude that, based on (1) analyses and test results on the retainer, (2) establishment and meeting of criteria for use of the device with ORAs modified for use with primary neutron sources in CR-3, (3) analyses which indicate that failure of the retainers, however unlikely, would not prevent plant safe shutdown and (4) failure detection capability of the LPMS, there is reasonable assurance that the proposed use of the retainer with two MORAs in CR-3 will pose no significant safety concern.

1.3 Inspection and Cleanup of Reactor Components

The inspection was accomplished using an underwater TV camera.(5) The camera was lowered both inside and outside the plenum cylinder, 360° around. A loose BPRA was discovered resting on one of the large plenum cylinder flow holes. Broken BPRA pins were also sighted, extending from the fuel assemblies. Debris was only sighted in the one quadrant. Prior to the plenum removal, the BPRA was extracted through the plenum cover. The plenum was lifted with no irregularities and placed on its storage stand in the deep end of the refueling canal. During the inspection described above, the only noticeable damage to the plenum was small marks on the lowest third of a flow hole in the plenum cylinder. No damage was noted on any of the control rod guide tubes.

After the plenum was removed, a fuel assembly was removed and a camera lowered into the lower part of the internals and vessel head. Several pieces of BPRA pins were spotted between the lower grid support forging and flow distribution plate. No damage to the internals was noted in this inspection.

Initial inspections of the core support assembly showed no damage; all eight of the vent valves were inspected. All vent valves, including the seating surfaces, were visually inspected with a TV camera. This inspection revealed no detrimental structural damage. The only indication of any type was a minor impact mark on one vent valve jack screw, believed to have occurred during removal of the plenum assembly. (Plenum assembly was removed from the vessel without the aid of the indexing fixture to facilitate removal of the BPRA assembly lodged in the plenum region.)

In addition to the detailed inspection, the vent valves were exercised and found to operate freely.

During the video inspection described above, several pieces of BPRA pins were spotted in the lower head region; the largest piece was approximately one foot long. No damage to the reactor vessel was noted.

The inspections of the reactor internals indicated no structural damage detrimental to the function of the reactor. The only damage attributable to the loose debris was some minor marks near a large flow hole in the plenum cylinder. This was believed to have been caused by impacting of the BPRA spider coupling before it escaped entirely from the fuel assembly. The fact that no other structural damage was found in the internals, although a significant amount of debris was found on the fuel assembly lower end fittings, the lower internals and the lower head of the reactor vessel, suggests that the parts that were able to pass through the system were too small to cause significant structural damage.

Documentation of inspection and cleanup operations was by video tape and independent observations by at least two observers. All debris observed using video equipment was removed by vacuuming and manual grabbers. Vacuuming removed debris from several inches in length down to debris that appeared as specks on the video screen. The manual grabbers removed debris from 12 feet long down to less than one inch.

In addition, an independent supply company representative was called to the Crystal River site to inspect the control rod drive lead screws and closure insert components. The results of this inspection, conducted under the reactor vessel head, indicated that no aluminum oxide debris was in the Control Rod Drive Mechanism (CRDM) internals. Further, it was pointed out by the licensee that:(5)

1. Inspection of CRDM components after design life testing have shown that a considerable amount of metallic debris could be present with no detrimental effect on mechanism operation.

2. Inspection of drives which have been ratchet tripped have shown that chips from the leadscrew can be present in the rotor assembly area of the mechanism. Presence of these chips has never prevented a control rod from being tripped or driven into the core.

Based on the above information, the licensee concluded that further CRDM inspection was not justifiable and that the CRDMs could continue in normal operation.

The above cleanup operations have resulted in approximately 402' out of the 403'8" of metal rod inventory from the two separated BRPAs being recovered from the fuel assemblies,(9) upper and lower end fittings, BPRA guide tubes, reactor vessel, core support assembly, plenum, and OTSGs. As described in the discussion on the OTSG, the seven plugged tubes in the "B" OTSG are suspected to contain some of the unrecovered debris. Both BPRA spiders and couplings were recovered; one intact in the plenum, one in pieces on OTSG B.

During the inspection, the licensee observed guide tube wear in two of the fuel assemblies. These assemblies are acceptable for further operation since no control components will be placed in these assemblies for the remainder of this cycle or in future cycles.(4)

Based on the above reported results, we agree that adequate measures have been taken to remove the BPRA debris. We also agree that an adequate inspection has been performed, thereby assuring no significant component degradation from the BPRA separations and failures and allowing continued operation. The effects during continued operation of any residual debris still contained in the reactor are further discussed in Section 1.4.

1.4 Residual Effects of BPRA Debris

The licensee evaluated the potential effects of residual poison and metallic fragments on CR-3 operational performance. In the analyses and evaluations, they conservatively assumed fuel assembly blockage conditions more severe than can be expected considering the amounts of debris recovered during the cleanup operations. Based on the results of these analyses and the unlikely occurrence of the assumed blockages, we agree that fuel failure resulting from BPRA debris is highly unlikely.

The licensee also concluded that there will be no adverse effects on the functions of the reactor internals as a result of the BPRA debris. Based on the results of the inspections and the cleanup operations, and that similar plants have operated for several months with similar (or greater) size debris, we agree with the licensee's conclusions.

The major area of concern is the potential for control rod binding as a result of any residual debris. The licensee has concluded that such a condition is remote based on the amount of metallic debris recovered, and the tortuous path required to produce interference sufficient to bind the control rods. The Technical Specification requirements on control rod insertion tests during startup tests provides additional assurance that there is no control rod binding.

Adverse effects on the reactor coolant pumps and RCP seals as a result of the BPRA debris have not been observed. The licensee has stated that pump vibration levels following the incident were comparable to the normal vibration levels prior to the incident. In addition, no evidence of loss of seal injection was observed. Since possible degradation of this type can be expected to be of a long term nature, the licensee will take the following action:

> The flow characteristics of the reactor coolant pumps will be verified by Performance Test Procedures prior to reactor criticality. This procedure establishes total core flow, loop flow mismatch, and flow coastdown (from a simultaneous trip of all four pumps) values for comparison to established acceptance criteria.
>
> The mechanical condition of each pump will be monitored when conducting Performance Test Procedures prior to reactor criticality and at approximately 25% power level increments following criticality. Through the use of permanently installed vibration instrumentation, frequency domain vibration signatures will be gathered for each pump. In addition, proximity probes, located at the pump to driver couplings, will be used to monitor shaft movement. Information is displayed on the Loose Parts Monitoring panel in the control room and alarms will sound if threshold vibration levels are exceeded.
>
> During normal plant operation, continuous vibration monitoring of the reactor coolant pumps is provided by the loose parts monitoring system. Automatic alarms trip when threshold values are exceeded. A requirement for daily functional checks of the vibration instrument system is planned for a future procedure revision.
>
> The integrity of the reactor coolant pump seal packages is verified by visual inspection through Surveillance Procedures prior to reactor criticality. Subsequent performance will be determined by monitoring seal package staging pressures and seal water leakoff temperatures.

We agree with the above conclusions and recommendations. In addition, should abnormal conditions occur, the anomalies will be reported to NRC within the requirements of the Technical Specifications.

We agree with the licensee in that the effects of the residual boron from the BPRAs in the coolant is insignificant as compared to the normal soluble boron levels. Therefore, no changes in boration procedures as a result of the BPRA failures are necessary

1.5 Steam Generator Damage and Repair

By letters dated June 8 and August 25, 1978, the licensee submitted a summary of the repairs of steam generator tube/tube sheet welds that were damaged as a result of the break-up of BPRAs. The licensee's repair program included video inspection and categorization of damaged tube stubs. leak testing, a 100% free path tube check, eddy current examinations (ECT), tube plugging, and dressing of the tube stubs.

In addition, a tube to tube sheet mock-up was prepared and tested at the B&W fabrication facility to verify the structural integrity of the damaged tube sheet area.

The tube stubs extend 0.3 inches above the upper tube sheet and the fillet seal welds extend about 0.1 inch above the tubesheet. Using video inspection, the damage to the tube stubs and seal welds in steam generator B was categorized as follows:(from least to most severe):

Class I (55% of the tubes)

Impact or roll over of the tube ends may exist on the O.D. or I.D. Deformed material does not include weld metal.

Class II (6% of the tubes)

Partially separated chip (sliver); may exist with Class I, III, or IV damage.

Class III (26% of the tubes)

Minor weld damage extending into the upper 1/3 of weld metal.

Class IV (17% of the tubes)

Damage to the tube ends and weld metal in excess of Class III. (Above percentages exceed 100% since Class II can exist with Class I, III, & IV).

Damage was in the form of cold working. No cracks were observed. Visual examination of steam generator A revealed no debris on the upper tubesheet, no tube end damage and no tube-to-tubesheet weld damage.

The leak tightness of the seal welds in both steam generators was verified by pressurizing the partially filled secondary side of the steam generator with helium and inspecting each weld individually with a mass spectrometer capable of detecting a 10^{-8} cc/sec leak. No leaks were observed.

A 100% free path check of all tubes in steam generators A and B was performed. Seven tubes in steam generator B, which had debris lodged in them that could not be removed, were plugged. Eddy current inspections of 3% of the tubes plus the 19 tubes from which debris was removed were conducted in steam generator B. Seven percent of the tubes in steam generator A were inspected, resulting in the plugging of one tube which had an ECT indication. No significant ECT indications other than those described were observed.

Dressing of the damaged tube stubs consisted of the removal of any metal slivers with hand tools. The licensee has performed flow calculations based on the reduced cross sections of the tube ends and has determined the effects to be negligible.

A ten tube mock-up of the tubesheet and damaged tube stubs was prepared by B&W. Hardness traverses across the damaged tube ends, welds, and into the clad showed the effects of significant cold working. Samples of the tube to tubesheet joints that were expanded, welded, and stress relieved per B&W fabrication procedures were tested with the welds completely removed. Results of these tests showed a minimum strength of 2500 pounds axial tube load to initiate motion of the tube relative to the tubesheet and a minimum load of 4520 pounds to completely free the tube from the tubesheet. The maximum axial load that the joint will experience during operation is 1100 pounds. The 23 inch thick tubesheet attenuates any lateral loads and their resulting moments.

Based on the above, we have determined that the licensee has conducted sufficient inspections of the damaged tube stubs, tubesheet, and steam generator tubes to discover any significant damage and adequate repairs have been completed. The leak tightness of the seal welds has been verified and the mechanical integrity of the expanded joint, which is not dependent on the seal welds, has been demonstrated. In addition, the CR-3 Technical Specifications currently impose a 1.0 gpm primary to secondary leak rate limit which will ensure that gradual degradation of the steam generator primary coolant boundary during operation will be detected. Therefore, we conclude that the licensee's steam generator repair program is acceptable and supports continued service of the steam generators.

The damage to the tube stubs and tubesheet was in the form of cold working which reduces the materials resistance to corrosion. If corrosion occurs, the corrosion rate in the primary coolant environment would be very slow and would not effect the tube or tubesheet integrity during the remainder of the current cycle of operation. However, during the next steam generator inspection a visual or video inspection should be conducted to verify that no detrimental corrosion effects have occurred. By letter dated August 31, 1978, FPC committed to conduct such an inspection.

1.6 Dropped Test Weight

As part of the CR-3 recovery program following failure of Burnable Poison Rod Assemblies, the reactor was defueled to allow inspection of fuel assemblies and reactor internals and retrieval of pieces of debris from the reactor coolant system. During conduct of maintenance activities on the fuel transfer mechanisms, a test weight device was inadvertently dropped which resulted in some damage to a fuel assembly located in the spent fuel pool.(7)

The damaging of one fuel assembly necessitated its removal and the removal of the three symmetrical assemblies, from APSR locations.(8) Four assemblies from another B&W core will be placed in locations containing Group 7 Regulating Rods and the assemblies they replace will be installed in the Axial Power Shaping Rod (APSR) locations. Because of the difference in the exposure of the APSR fuel assemblies, new limits were placed on APSR insertion so that power peaking limits are not exceeded. The insertion limit is based on loss-of-coolant accident (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. The proposed Technical Specifications implement those limits.

The APSR that was located in the impacted fuel assembly was also inspected for damage.(9) In addition to a visual inspection, a frictional pull test was performed on the APSR. Based on the results of these inspections and test, the licensee determined that the APSR could be returned to service.

Since the damaged fuel assembly will not be used it will not affect core performance during the remainder of this cycle. Sections 1.7 and 1.8 address our review on the four replacement assemblies.

We agree that the inspection and pull test on the APSR provides sufficient confidence that the APSR will adequately perform its intended function.

1.7 Nuclear Design

The nuclear design of the reactor core will be altered by removing the BPRAs and ORAs and by substituting four fuel assemblies from another reactor to symmetrically replace an assembly which was damaged during fuel handling.(4) The effects of these changes are relatively small because the BPRAs were largely depleted of ^{10}B by shutdown and because the enrichment of the replacement assemblies were chosen such that effect would be minimized.

Removal of the BPRAs primarily results in changes in local peaking in the fuel assemblies where they had resided. (The ORAs had no neutronic function and their removal has essentially no effect on the nuclear design.) In addition, some reactivity worth remains even at 269 EFPD. Therefore, removal will increase core reactivity and modify the gross power distribution. Control rod worths and part-length rod worths will change, and all kinetics parameters will be perturbed.

The licensee has accounted for all these effects by reanalyzing the remainder of the cycle as if it were a totally new cycle, using the same calculational tools used in the original analysis. The discrepancy in the exposure tallies (and thus in the isotopic inventories) in the two fuel assemblies from which the burnable poison clusters were ejected was estimated and found to be less than 0.4% in exposure. This is smaller than the normal uncertainty in exposure tallies. Therefore, no special allowance was made for this discrepancy.

Replacement of the four quadrant symmetric assemblies resulted in a core which is quadrant symmetric rather than octant symmetric. Therefore, the calculations were based upon quadrant symmetry, although the resulting power distributions are nearly octant symmetric. The symmetry assumptions, as well as the power distributions and certain rod worths, will be verified during the startup program. We find these calculations to be appropriate and acceptable.

1.8 Thermal-Hydraulic Design

Removal of the BPRAs and ORAs will allow more coolant to flow through the open guide tubes in the fuel assemblies. This will increase the maximum bypass flow rate from 6.04% to 10.40%. (4) To offset this, the licensee has proposed to reduce the Technical Specification limit on $F_{\Delta H}$ from 1.78 to 1.71. Nuclear calculations predict a maximum $F_{\Delta H}$ of 1.596 during the modified cycle, so there should be no difficulty meeting the requirement. We find this acceptable.

The four replacement fuel assemblies are of the older Mark B2 rather than the newer Mark B3 design. These assemblies have slightly higher flow resistance. However, the nuclear design calculations predict a maximum axially integrated heat rate in these assemblies that is 60% below the design value. Thus, these assemblies will never be limiting. We find this acceptable.

1.9 Accident and Transient Analyses

The modified core will have slightly modified kinetics parameters (Section 1.7). The licensee has compared the calculated parameters to those assumed in the Final Safety Analysis Report (FSAR) accident and transient analyses and found that the FSAR values bound all the revised parameters with the exception of the revised Doppler coefficient, which is more negative than the FSAR range.

A more negative Doppler coefficient improves core response except in the case of "cool-down" incidents. Therefore, the FSAR analyses remain valid except for incidents initiated by a decreasing temperature.

The "cool-down" incidents are bounded by the Steam Line Break at end-of-cycle. Although the change in Doppler coefficient is in the non-conservative direction for this event, its effect is second-order with respect to the moderator temperature coefficient, which has changed in the opposite direction. Because the effect of the change in the moderator coefficient is expected to be at least a factor of 16 greater than that of the Doppler coefficient, we agree that the results of the FSAR analysis remain valid. Therefore, we find the analyses of the accidents and transients to be acceptable.

The analysis of the LOCA is not affected by the removal of the BPRAs and ORAs. Technical Specification limits on control rod and axial power shaping rod positions, coupled with limits on axial imbalance and tilt, will assure operation within the range of initial conditions assumed in the FSAR LOCA analyses. Therefore, the FSAR analyses remain applicable. Recent concerns regarding B&W's small break analyses are addressed in the Exemption which accompanies this evaluation.

1.10 Physics Startup Test

The physics startup test program(4) has been reviewed. Additional information was requested and supplied in Reference 9. The physics startup test program includes zero power measurements of critical boron concentration, temperature coefficients, ejected control rod worth and control rod group reactivity worth. Power distribution, temperature coefficient and power coefficient measurements will be made at higher powers. The acceptance criteria and the actions to be taken if the acceptance criteria are not met were reviewed as well as the tests. The licensee has stated that the action to be taken if the sum of the worth of groups 5, 6, and 7 differs from the predicted by more than $\pm 10\%$, is to measure group 4 and that if the sum of the worths of groups 4, 5, 6, and 7 differs from the predicted by more than $\pm 10\%$, a complete safety evaluation of the discrepancy will be made.

It should be noted that during zero power testing of ejected rod worth, four symmetric rods will be tested. This will provide additional verification of quadrant symmetry.

A summary of the results of this test program will be submitted to the NRC within 90 days after completion of the program.

This entire program has been reviewed by the NRC staff and found to be acceptable.

1.11 Technical Specifications

The licensee has proposed changes to the Technical Specifications as follows:

- A reduced $F_{\Delta H}$ limit and modified pressure/temperature limits to accommodate the increased bypass flow resulting from BPRA and ORA removal.
- An increased minimum boric acid storage inventory to accommodate the effect of BPRA removal on the capability to override xenon and come to cold shutdown.
- Modified control rod insertion limits, to preserve shutdown margin at hot shutdown conditions.
- Modified APSR insertion limits, due to the fuel shuffle in the APSR locations.
- New axial power imbalance envelopes to assure the linear heat generation rate and operating DNBR limits assumed in the accident and transient analyses.

We have reviewed the changes to the Technical Specifications and found them to be acceptable.

1.12 Conclusion

Based on our evaluation of the application for amendment and available information, we find the proposed changes to the Technical Specifications to be acceptable, and conclude that it is acceptable for the licensee to proceed with modified Cycle 1 operation in the manner proposed.

2.0 Quadrant Power Tilt Limits

Certain maximum allowable values of neutron flux tilt have been established for the CR-3 reactor core. These limits were established by B&W, to assure that the radial power distribution satisfies the design values used in the power capability analyses, and to monitor for anomalous behavior in the core.

To allow for the uncertainty associated with these measurements, B&W had estimated the magnitude of the uncertainty for various types and conditions of measurements and established a maximum allowable measured value for flux tilt. These allowable measured values are smaller than the allowable actual values (also called measurement independent values) by the amount of the uncertainty. The allowable measured values also vary with the type and extent of instrumentation used for the measurement.

The allowable values of measured flux tilt previously in use were based on an error analysis performed by B&W in 1974 using data obtained with prototype detectors. Operating experience since that time, however, had indicated that the instrumental uncertainties might not be sufficiently conservative, and therefore, there was a need for a reevaluation of these uncertainties. Such a reevaluation program was initiated by B&W early in 1978 and a report describing the program and its results was transmitted to the NRC staff by B&W letter of May 11, 1978.

We have reviewed the B&W report of May 11, 1978, on in-core detector measurement errors. The report considers the observed uncertainties associated with the various types of detectors in use and the effect of detector neutron exposure on the uncertainty. The report also describes the error propagation and statistical analyses that were performed to develop conservative uncertainty corrections for each type of detector as a function of neutron exposure. Based on the analyses, the report recommends new, more restrictive values of maximum allowable measured flux tilt for various measurement techniques.

Based on our review of the B&W report, we have concluded that the analytical methods are acceptable. We have also reviewed the recommended maximum allowable measured flux tilt setpoints applicable to CR-3 and conclude that these recommended values are also acceptable. Since the numerical changes to the Technical Specifications requested by the licensee follow the B&W recommendations, we have concluded that the requested numerical changes are likewise acceptable.

The licensee has also requested that the measurement independent values be deleted from the Limiting Conditions for Operation (LCO) and added to the bases. This involves a change to the B&W Standard Technical

Specification format and as such, will be considered separately.

3.0 Atypical Weld Wire and Pressure-Temperature Operating Limits

By letter dated August 18, 1978, the licensee submitted a set of revised pressure-temperature operating curves to be applied to CR-3 administratively. The proposed operating curves were calculated for operation through 5 effective full power years (EFPY). There are seven weld materials in the reactor vessel beltline region. Based on chemical composition and weld location, WF-70 is expected to be the most limiting vessel material. The licensee's submittal of August 18, 1978, included revisions to reflect the possible use of atypical weld wire in weld WF-70. This atypical wire had a lower nickel content (0.1% vs typically 0.6%) and a higher silicon content (1% vs typically 0.5%) than the wire normally used. The effect of this variation is to cause a higher initial reference temperature for the nil ductility transition (RT_{NDT}). This higher value of RT_{NDT} was used by the licensee in his analysis.

We have reviewed the licensee's submittal. Based on our review we agree that weld WF-70 is the limiting material. We estimate that the maximum value of fluence on this weld at the 1/4 T location at end of life will be 1.55×10^{18} n/cm^2. Using this fluence value and Regulatory Guide 1.99, Revision 1 to predict radiation damage, we calculate the proposed operating curves are acceptable for operation through 3 EFPY. For operation through 3 EFPY the proposed pressure-temperature operating limits are in conformance with Appendix G, 10 CFR Part 50.

The licensee has proposed to administratively apply the revised pressure-temperature limits until (1) it is determined that the atypical weld wire was not used, or (2) until a license amendment is proposed and issued that inserts the revised limits into the Technical Specifications. CR-3 has only been operated for about 270 effective full power days. Since the proposed curves are good for 3 EFPY they are applicable now and for a few more years.

Because it is not known that the atypical weld wire was used in the CR-3 reactor vessel and the current record search by B&W may produce evidence that this weld wire was not used, we have determined that applying these limits administratively for an interim period is acceptable. By letter dated September , 1978, the licensee has committed to propose revised pressure-temperature Technical Specifications as soon as the atypical weld material is determined to have been used, or by November 1, 1978, if it has not, by that time, been determined. We find this 2 month interval an acceptable maximum interim period to administratively apply these limits.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: September 1, 1978

REFERENCES

1. Letter, A. J. Ormston (FPC) to Director, Office of Nuclear Reactor Regulation (NRC) dated June 6, 1978.

2. Letter, W. P. Stewart (FPC) to Director, Office of Nuclear Reactor Regulation (NRC) dated July 21, 1978.

3. Letter, W. P. Stewart (FPC) to J. P. O'Reilly (NRC) dated March 17, 1978, enclosing Licensee Event Report 78-017/01T-0.

4. Letter, W. P. Stewart (FPC) to R. W. Reid (NRC) dated July 21, 1978, enclosing "Crystal River Unit 3 Licensing Considerations for Continued Operation Without Burnable Poison Rod Assemblies and Orifice Rod Assemblies," BAW-1490, Rev. 1, July 1978.

5. Letter, W. P. Stewart (FPC) to R. W. Reid (NRC) dated May 16, 1978.

6. Letter, J. H. Taylor (B&W) to S. A. Varga (NRC) dated June 7, 1978, enclosing "BPRA Retainer Design Report," BAW-1496, May 1978.

7. Letter. W. P. Stewart (FPC) to J. P. O'Reilly (NRC) dated June 22, 1978, enclosing Licensee Event Report 78-031/01T-0.

8. Letter, W. P. Stewart (FPC) to Director, Office of Nuclear Reactor Regulation (NRC) dated June 28, 1978.

9. Letter, W. P. Stewart (FPC) to R. W. Reid (NRC) dated August 22, 1978.