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Docket No. 50-302

Mr. John A. Hancock
Vice President, Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Licensing
P. O. Box 14042, M.A.C. H-2
St. Petersburg, Florida 33733

Dear Mr. Hancock:

The Commission has issued the enclosed Amendment No. 55 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The amendment consists of changes to the Technical Specifications (TSs) in response to your requests dated March 4, 1982, as supplemented March 9, 1982; April 1, 1982, as supplemented April 6, 1982, and April 30, 1982. The amendment formalizes and supplements those changes previously authorized by telephone on March 3, 1982, March 9, 1982, April 1, 1982 and April 6, 1982, and by confirmatory letters dated March 12, 1982, April 2, 1982, and April 16, 1982.

The amendment (1) revises the response time of the Reactor Coolant Pump Power Monitors (RCPPMs), (2) allows operation of CR-3 at a power level no greater than 2300 MWt (90.4% of full power) with the RCPPMs bypassed, and (3) administratively adds limiting conditions for operation (LCO) and surveillance requirements for the power operated relief valves (PORV) which had inadvertently been omitted from a previous amendment. Changes proposed by your April 9, 1982, application with regard to three pump operation with the RCPPMs bypassed have not been approved because you did not provide sufficient information for our review.

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

Sincerely,

Original signed by

Sydney Miner, Sr. Project Manager
Operating Reactors Branch #4
Division of Licensing

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PDR ADOCK 05000302
P PDR

Enclosures:
1. Amendment No. 55
2. Safety Evaluation
3. Notice

| OFFICE | cc w/enclosures: | | ORB#4:DL | ORB#4:DL | C-ORB#4:DL | ~~AD-OR:DL~~ | OELD |
|---|---|---|---|---|---|---|---|
| SURNAME | See next page | | RIngram | SMiner/cab | JStolz | TNovak | M Karman |
| DATE | | | 7/14/82 | 7/14/82 | 7/14/82 | 7/12/82 | 7/15/82 |

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

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Vice President and General Counsel
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St. Petersburg, Florida 33733

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Board of County Commissioners
Citrus County
Iverness, Florida 36250

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Power Plant Siting Section
State of Florida
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Tallahassee, Florida 32301

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Mr. James P. O'Reilly, Regional Administrator
U. S. Nuclear Regulatory Commission, Region II
101 Marietta Street, Suite 3100
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER CORPORATION
CITY OF ALACHUA
CITY OF BUSHNELL
CITY OF GAINESVILLE
CITY OF KISSIMMEE
CITY OF LEESBURG
CITY OF NEW SMYRNA BEACH AND UTILITIES COMMISSION, CITY OF NEW SMYRNA BEACH
CITY OF OCALA
ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO
SEBRING UTILITIES COMMISSION
SEMINOLE ELECTRIC COOPERATIVE, INC.
CITY OF TALLAHASSEE

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 55
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 March 4, 1982, and April 1, 1982, as supplemented by letters dated March 9, 1982, April 6, 1982, and April 30, 1982, 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. 55, are hereby incorporated in the license. Florida Power Corporation shall operate the facility in accordance with the Technical Specifications.

3. Portions of this license amendment became effective March 3, 1982, March 9, 1982, April 1, 1982, and April 6, 1982, and portions of this amendment are effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 15, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 55

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

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

2-5

2-6

2-7

B2-4

B2-5

B2-7

3/4 3-2

3/4 3-3

3/4 3-4

3/4 3-5

3/4 3-6

3/4 4-4

3/4 4-4a (new page)

## TABLE 2.2-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| | FUNCTIONAL UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|---|
| 1. | Manual Reactor Trip | Not Applicable | Not Applicable |
| 2. | Nuclear Overpower | $\leq$ 94.8% of RATED THERMAL POWER with four pumps operating<br><br>$\leq$ 79.92% of RATED THERMAL POWER with three pumps operating (3) | $\leq$ 94.8% of RATED THERMAL POWER with four pumps operating<br><br>$\leq$ 79.92% of RATED THERMAL POWER with three pumps operating (3) |
| 3. | RCS Outlet Temperature - High | $\leq$ 618°F | $\leq$ 618°F |
| 4. | Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE (1) | Trip Setpoint not to exceed the limit line of Figure 2.2-1 | Allowable Values not to exceed the limit line of Figure 2.2-1 |
| 5. | RCS Pressure - Low (1) | $\geq$ 1800 psig | $\geq$ 1800 psig |
| 6. | RCS Pressure - High | $\leq$ 2300 psig | $\leq$ 2300 psig |
| 7. | RCS Pressure - Variable Low (1) | $\geq$ (11.59 $T_{out}$ °F - 5037.8) psig | $\geq$ (11.59 $T_{out}$ °F - 5037.8) psig |

TABLE 2.2-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION TRIP SETPOINTS

| | FUNCTION UNIT | TRIP SETPOINT | ALLOWABLE VALUES |
|---|---|---|---|
| 8. | Pump Status Based on Reactor Coolant Pump Power Monitors (1) (2) | More than one pump drawing $\leq$ 3900 kw or $\geq$ 9000 kw | More than one pump drawing $\leq$ 3900 kw or $\geq$ 9000 kw |
| 9. | Reactor Containment Vessel Pressure High | $\leq$ 4 psig | $\leq$ 4 psig |

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

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

(2) Trip may be manually bypassed when reactor power is less than or equal to 2300 MWt.

(3) Operation with three reactor coolant pumps with RCPPMs bypassed is not permitted.

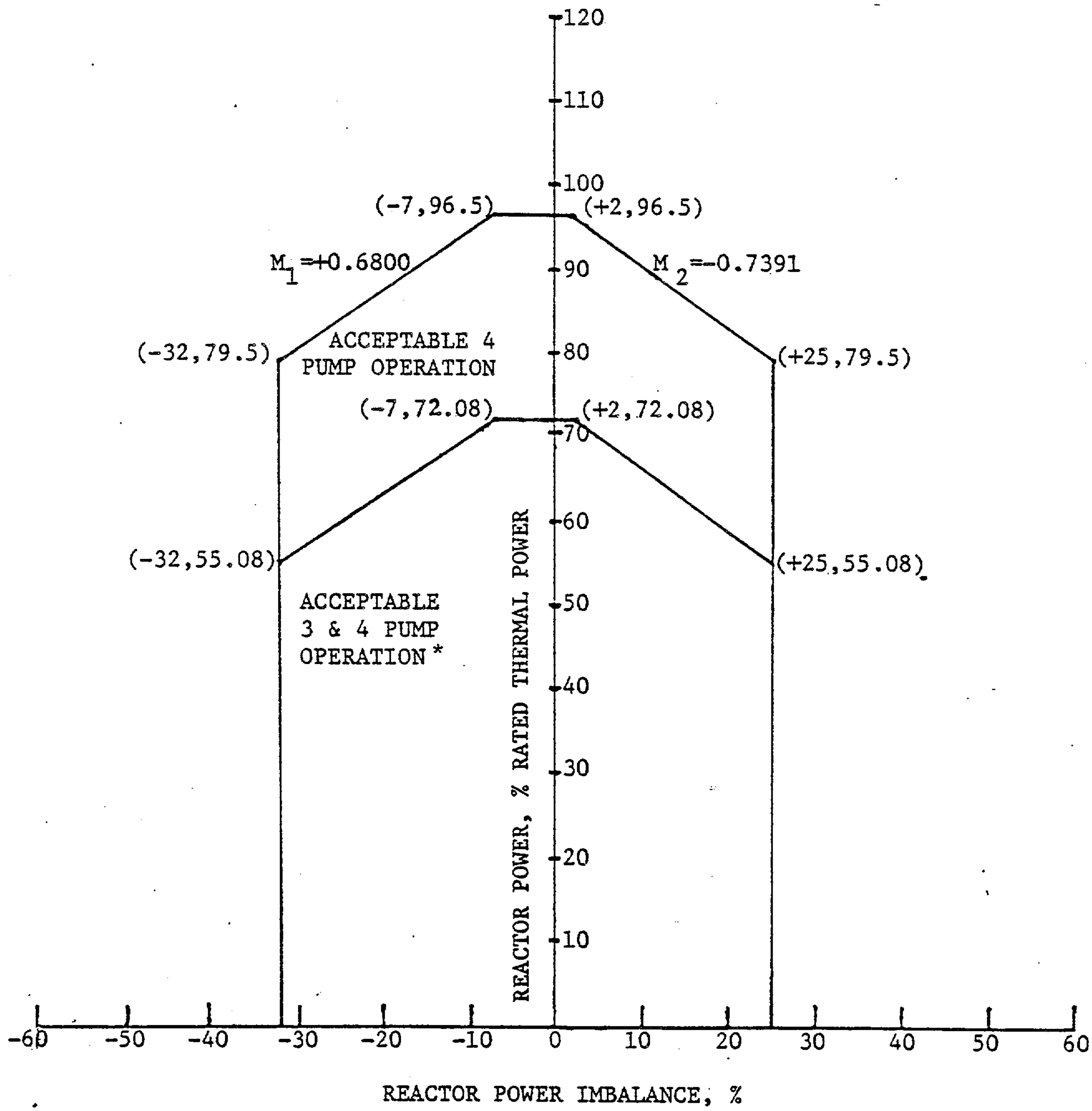



FIGURE 2.2-1

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

*Operation with three reactor coolant pumps with RCPPMs bypassed is not permitted.

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 three pump operation is more restrictive than any other reactor coolant pump situation because any pressure/temperature point above and to the left of the three pump curve will be above and to the left of the other curves.

## 2.1.3 REACTOR COOLANT SYSTEM PRESSURE

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

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

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

## 2.2.1 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 94.8% of rated power. Due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 101.9% which was used in the safety analysis.

RCS Outlet Temperature - High

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

Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE

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

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

1. Trip would occur when four reactor coolant pumps are operating if power is $\geq$ 96.5% and reactor flow rate is 100%, or flow rate is $\leq$ 93.69% and power level is 90.41%.

2. Trip would occur when three reactor coolant pumps are operating if power is $\geq$ 79.92% and reactor flow rate is 74.7%, or flow rate is $\leq$ 70.09% and power is 75%. Operation with three reactor coolant pumps with RCPPM's bypassed is not permitted.

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

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

## RCS Pressure - Low, High, and Variable Low

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

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

The RCS Pressure-Low, 1800 psig, and RCS Pressure-Variable Low, ($11.59\ T_{out}$ °F - 5037.8) psig, Trip Setpoints have been established to maintain the DNB ratio greater than or equal to 1.30 for those design accidents that result in a pressure reduction. It also prevents reactor operation at pressures below the valid range of DNB correlation limits, protecting against DNB.

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

## BASES

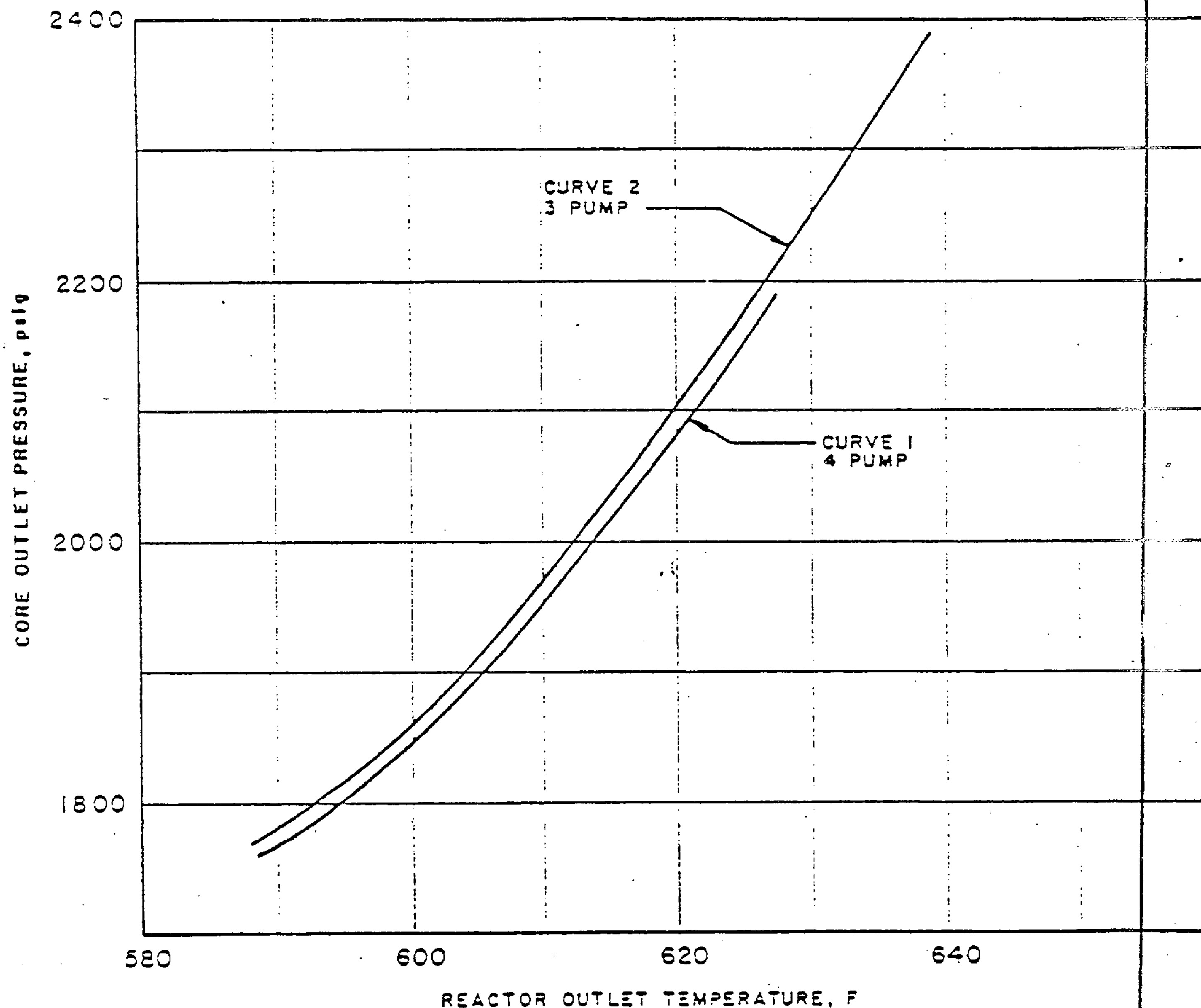
### Reactor Containment Vessel Pressure - High

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

### Reactor Coolant Pump Power Monitors

In conjunction with the power/imbalance/flow trips, the Reactor Coolant Pump Power Monitors trip prevents the minimum core DNBR from decreasing below 1.30 by tripping the reactor due to more than one reactor coolant pump not operating.

A reactor coolant pump is considered to be not operating when the power required by the pump is $\geq$ 120% or is $\leq$ 70% of the nominal operating power. The nominal operating power decreases from when a pump is first started during heatup and is pumping dense fluid (typically 7500 KW) to when a pump is operating at full reactor power and is pumping less dense fluid (typically 5500KW). In order to avoid spurious trips during normal operation, the 120% trip setpoint (9000KW) is based on the nominal operating power for a pump during heatup and the 70% trip setpoint (3900KW) is based on the nominal operating power for a pump operating at full reactor power. Florida Power has agreed to take credit for the pump overpower trip in order to assure that certain potential faults such as a seismically induced fault high signal will not prevent this instrumentation from providing the protective action (i.e., a trip signal).




REACTOR COOLANT FLOW

| CURVE | FLOW (% DESIGN) | POWER (RTP) | PUMPS OPERATING (TYPE OF LIMIT) |
|---|---|---|---|
| 1 | $139.7 \times 10^6$ (106.5%) | 113.05% | 4 PUMPS (DNBR) |
| 2 | $104.4 \times 10^6$ (79.6%) | 90.84% | 3 PUMPS (DNBR) |

PRESSURE/TEMPERATURE LIMITS AT MAXIMUM
ALLOWABLE POWER FOR MINIMUM DNBR

BASES FIGURE 2.1

## 3/4.3 INSTRUMENTATION

### 3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

#### LIMITING CONDITION FOR OPERATION

3.3.1.1 As a minimum, the Reactor Protection System instrumentation channels and bypasses of Table 3.3-1 shall be OPERABLE with RESPONSE TIMES as shown in Table 3.3-2.

APPLICABILITY: As shown in Table 3.3-1.

ACTION:

As shown in Table 3.3-1.

#### SURVEILLANCE REQUIREMENTS

4.3.1.1.1 Each Reactor Protection System instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST operations during the MODES and at the frequencies shown in Table 4.3-1.

4.3.1.1.2 The total bypass function shall be demonstrated OPERABLE at least once per 18 months during CHANNEL CALIBRATION testing of each channel affected by bypass operation.

4.3.1.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip function shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per function such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip function as shown in the "Total No. of Channels" column of Table 3.3-1.

# TABLE 3.3-1

## REACTOR PROTECTION SYSTEM INSTRUMENTATION

| | FUNCTIONAL UNIT | TOTAL NO. OF CHANNELS | CHANNELS TO TRIP | MINIMUM CHANNELS OPERABLE | APPLICABLE MODES | ACTION |
|---|---|---|---|---|---|---|
| 1. | Manual Reactor Trip | 1 | 1 | 1 | 1, 2 and * | 8 |
| 2. | Nuclear Overpower | 4 | 2 | 3 | 1, 2 | 2# |
| 3. | RCS Outlet Temperature - High | 4 | 2 | 3 | 1, 2 | 2# |
| 4. | Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE | 4 | 2(a) | 3 | 1, 2 | 2# |
| 5. | RCS Pressure - Low | 4 | 2(a) | 3 | 1, 2 | 3# |
| 6. | RCS Pressure - High | 4 | 2 | 3 | 1, 2 | 3# |
| 7. | Variable Low RCS Pressure | 4 | 2(a) | 3 | 1, 2 | 3# |
| 8. | Reactor Containment Pressure - High | 4 | 2 | 3 | 1, 2 | 3# |
| 9. | Intermediate Range, Neutron Flux and Rate | 2 | 0 | 2 | 1, 2 and * | 4 |
| 10. | Source Range, Neutron Flux and Rate | | | | | |
| | A. Startup | 2 | 0 | 2 | 2## and * | 5 |
| | B. Shutdown | 2 | 0 | 1 | 3, 4 and 5 | |
| 11. | Control Rod Drive Trip Breakers | 2 per trip system | 1 per trip system | 2 per trip system | 1, 2 and * | 7# |
| 12. | Reactor Trip Module | 2 per trip system | 1 per trip system | 2 per trip system | 1, 2 and * | 7# |
| 13. | Shutdown Bypass RCS Pressure - High | 4 | 2 | 3 | 2**, 3**, 4**, 5** | 6# |
| 14. | Reactor Coolant Pump Power Monitors | 2 per pump | 1 from 2 or more pumps (a,b) | 2 per pump | 1, 2 | 25 |

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the control rod drive trip breakers in the closed position and the control rod drive system capable of rod withdrawal.

**When Shutdown Bypass is actuated.

#The provisions of Specification 3.0.4 are not applicable.

##High voltage to detector may be de-energized above $10^{-10}$ amps on both Intermediate Range channels.

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

  (1) The Nuclear Overpower Trip Setpoint is $\leq$ 5% of RATED THERMAL POWER,

  (2) The Shutdown Bypass RCS Pressure - High Trip Setpoint of $\leq$ 1720 psig is imposed, and

  (3) The Shutdown Bypass is removed when RCS pressure >1800 psig.

(b) Trip may be manually bypassed when reactor power is less than or equal to 2300 MWt and 4 reactor coolant pumps are operating.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and/or open the control rod drive trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided all of the following conditions are satisfied:

  a. The inoperable channel is placed in the tripped condition within one hour.

  b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.

TABLE 3.3-1 (Continued)

ACTION STATEMENTS (Continued)

and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1, and

c. Either, THERMAL POWER is restricted to ≤ 75% of RATED RATED THERMAL and the Nuclear Overpower Trip Setpoint is reduced to ≤ 85% of RATED THERMAL POWER within 4 hours or the QUADRANT POWER TILT is monitored at least once per 12 hours.

ACTION 3 - With the number of OPERABLE channels one less than the Total Number of Channels STARTUP and POWER OPERATION may proceed provided both of the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within one hour.

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1.

Action 4 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement and with the THERMAL Power level:

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

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

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

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

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

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

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

a. Within 1 hour:

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

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

b. One additional channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1, and the inoperable channel above may be bypassed for up to 30 minutes in any 24 hour period when necessary to test the trip breaker associated with the logic of the channel being tested per Specification 4.3.1.1. The inoperable channel above may not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.

ACTION 8 - With the number of channels OPERABLE less than required by the Minimum Channels OPERABLE requirement, be in at least HOT STANDBY within 6 hours.

ACTION 25 - With the number of channels OPERABLE one less than the required Minimum Channels OPERABLE requirement, plant operation may continue until the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 4 hours.

## TABLE 3.3-2

## REACTOR PROTECTION SYSTEM INSTRUMENTATION RESPONSE TIMES

| | Functional Unit | Response Times |
|---|---|---|
| 1. | Manual Reactor Trip | Not Applicable |
| 2. | Nuclear Overpower * | $\leq$ 0.266 seconds |
| 3. | RCS Outlet Temperature - High | Not Applicable |
| 4. | Nuclear Overpower Based on RCS Flow and AXIAL POWER IMBALANCE * | $\leq$ 1.79 seconds |
| 5. | RCS Pressure - Low | $\leq$ 0.44 seconds |
| 6. | RCS Pressure - High | $\leq$ 0.44 seconds |
| 7. | Variable Low RCS Pressure | Not Applicable |
| 8. | Pump Status Based on RCPPMs | $\leq$ 0.56 seconds |
| 9. | Reactor Containment Pressure - High | Not Applicable |

* Neutron detectors are exempt from response time testing. Response time of the neutron flux signal portion of the channel shall be measured from detector output or input of first electronic component in channel.

## RELIEF VALVES - SHUTDOWN

### CODE SAFETY VALVES

LIMITING CONDITION FOR OPERATION

---

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psig ± 1%.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE DHR loop into operation.

SURVEILLANCE REQUIREMENTS

---

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



APR 17 1981

## RELIEF VALVES - OPERATING

## CODE SAFETY VALVES

### LIMITING CONDITION FOR OPERATION

3.4.3.1 All pressurizer code safety valves shall be OPERABLE with a lift setting of 2500 psig $\pm$ 1%.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.

### SURVEILLANCE REQUIREMENTS

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

REACTOR COOLANT SYSTEM

POWER OPERATED RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.3.2 The power operated relief valve (PORV) and its associated block valve shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

a. With the PORV inoperable, within 1 hour either restore the PORV to OPERABLE status or close the associated block valve and remove power from the block valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With the block valve inoperable, within 1 hour either restore the block valve to OPERABLE status or close the block valve and remove power from the block valve or close the PORV and remove power from the associated solenoid valve; otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 In addition to the requirements of Specifications 4.0.5, the PORV shall be demonstrated OPERABLE at least once per 18 months by performance of a CHANNEL CALIBRATION.

4.4.3.2.2 The block valve shall be demonstrated OPERABLE at least once per 92 days by operating the valve through one complete cycle of full travel.

4.4.3.2.3 The emergency power supply for the PORV and block valve shall be demonstrated OPERABLE at least once per 18 months by transferring motive and control power from the normal to the emergency power supply and operating the valve through a complete cycle of full travel.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 55 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

## Introduction

Amendment No. 41 to Facility Operating License No. DPR-72 (Reference 1) authorized Florida Power Corporation (the licensee or FPC) to operate Crystal River Unit No. 3 (CR-3) at a power level of 2544 MWt, increased from the previously authorized level of 2452 MWt. As part of the review of this increased power request, the NRC determined that the time response of the previous flux-to-flow monitoring instrumentation was too slow to avoid violating the Departure from Nucleate Boiling Ratio (DNBR) criterion during certain postulated loss-of-flow transients. The licensee installed a pump power monitoring (PM) instrumentation system which could detect loss of power to more than one reactor coolant pump (RCP) on a much faster basis (620 ms vs. 1400 ms). Since then, plant operation has been interrupted several times due to spurious operation of the RCPPMs.

To resolve the spurious trip problem, the licensee proposed increasing the response time of the RCPPMs by 100 to 150 msec. By telephone the evening of March 3, 1982, and confirmed by letter dated March 4, 1982, the licensee requested authorization to operate the CR-3 plant at no more than 75% of full power with response times greater than those in the Technical Specifications (TSs) until they could justify a proposed new response time. The authorization was granted on March 3, 1982. By letter dated March 9, 1982, the licensee provided adequate justification for a proposed new response time for the RCPPMs. On March 9, 1982, we authorized operation at 100% power with the RCPPM response time increased from $\leq$0.47 seconds to $\leq$0.56 seconds.

On March 26, 1982, CR-3 was again tripped by its RCPPMs. Increasing the response time to 560 msec did not resolve the spurious trip problem. Therefore, in a letter dated April 1, 1982, the licensee proposed operating at less than 100% power with the RCPPMs bypassed until they could establish the cause of the PM trips and implement a fix. They further stated that they expected to be able to justify operation at 90% of full power with the RCPPMs bypassed. However, until they could provide the justification, they would limit reactor power to 75% of full power.

On April 1, 1982, we authorized operation at 75% of full power with bypassed RCPPMs until operation at 90.4% full power (2300 MWt) could be justified. Subsequently, by letter dated April 6, 1982, the licensee provided the necessary justification for operation of CR-3 at 2300 MWt (90.4% of full power), and we approved operation at 90.4% full power with the RCPPMs bypassed. By letter dated April 9, 1982, the licensee requested approval for operation at 75% of 2300 MWt for three RCPs with the RCPPMs bypassed. Subsequently, by letter dated April 30, 1982, the licensee submitted a TS change request supplementing the information provided on April 1 and April 6. However,

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neither the April 9, 1982 or April 30, 1982 submittals contained suitable analyses related to plant operation with only three RCPs in operation.

Discussion & Evaluation

1. Operation at 75% and 90.4% Power With The RCPPMs Bypassed

(a) Discussion

The licensee initially indicated that for the present core operation at about 90% of full power with the RCPPMs bypassed, the flux-to-flow trip is adequate to prevent violating the DNBR limit. However, since they did not provide justification for this power, operation was limited to 75% of full power. Subsequently, FPC submitted a summary of a Babcock & Wilcox Company (B&W) analysis of the four pump loss of coolant flow without taking credit for the RCPPM. This analysis relied on the flux/flow trip and shows that the minimum DNBR is 1.43 for a four pump loss of flow initiated from 2300 MWt (90.4% of full power).

(b) Evaluation

The effect of bypassing the RCPPM trip can be balanced by reducing reactor power. Operation at 75% of full power with the RCPPMs bypassed is acceptable since the existing flux/flow trip setpoint and response times adequately compensate for operation without the RCPPMs at this power level.

The analysis of the four pump loss of flow at 90.4% full power was done with the same codes, methods and correlation as previously used in CR-3 licensing analyses. The analyses indicate that the minimum DNBR would be 1.43 which is above the DNBR limit of 1.35. The analyses therefore demonstrate acceptable results.

B&W was contacted to assure that the analyses had been performed in conformance with B&W quality assurance procedures. Technical aspects of the analyses were also discussed to assure that the results were reasonable and consistent with previous analyses.

Sufficient analyses or discussions relative to three pump operation were not included in the licensee's submittal.

Based on the results of the analyses submittal by FPC, we conclude that a four pump loss of coolant flow event initiated at 2300 MWt (90.4% of full power) would not exceed the applicable regulatory limits, therefore approval of the proposed TS, as it relates to four pump operation with RCPPMs bypassed, will not result in an undue risk to the health and safety of the public. The proposed changes related to three pump operation are not acceptable since the licensee did not provide suitable analyses to justify the proposed changes.

2. RCPPMs Response Times

(a) Discussion

The RCPPMs protect against a multiple-pump loss-of-flow transient by tripping the reactor early enough to prevent exceeding the DNBR criterion of 1.30

minimum. This is accomplished by causing control rod insertion (scram) to start within 620 ms of the occurrence of improper pump power conditions.

When the original authorization for the RCPPMs was issued, the time response was allocated as: 240 ms for the sensor channel (sensor elements, watt transducer, bistable, and output relay); 150 ms for the Reactor Protection System (RPS) (logic, etc.); 80 ms for Control Rod Drive (CRD) breakers; 90 ms allocated for uncertainty (margin); and 60 ms for release of the control rods (roller nuts from lead screws).

Subsequently, the licensee's request (Reference 2) for TS changes stated that "Recent revisions of the RPS instrumentation delay times are the result of an instrument error analysis by B&W." Accordingly, a later amendment (Reference 3) showed the response time for the RCPPMs to be 470 ms. When the intentional time delay (100-150 ms) was added, attaining the 470 ms limit became questionable.

At the same time, the question arose about the adequacy of the time response tests being conducted to demonstrate compliance with the facility TSs. The tests did not conform in an obvious way to the definition of RPS response times in that the sensor elements and the "watt transducers" were not being tested. The licensee contended that use of manufacturer's data was sufficient.

(b) Evaluation

The original design allocations of response times were listed above. The crucial point of these values is that the plant transient analysis was run under the assumption of a 620 ms delay to the start of negative reactivity insertion (rod motion). While the 620 ms assumption remains valid, several factors have been considered and are discussed below.

1. Optical isolators were installed between the RCPPM output relays and the RPS to prevent 6.9 kv from being applied to the RPS by certain postulated failures. The design allocation of response time was 150 ms. The original design allocation of 240 ms for the sensor channel was refined as follows: 16 ms for the sensor elements (current transformers and potential transformers); 72 ms for the watt transducer, bistable, and output relay; and 150 ms for the optical isolator.

2. The design allocation for the RPS (logics, etc.) was revised from 150 ms to 131 ms. The basis for this revision is extensive testing associated with equipment qualification programs and is documented in B&W Topical Report BAW-1003. This revision increased the "dedicated margin" for uncertainty from 90 ms to 109 ms.

3. Due to some operational problems, the operating voltage of the isolators had to be increased. Type testing of the isolators at the new operating voltage shows that at an ambient temperature of 65°F the isolators responded in 106 ms or less. At higher temperatures the response is faster. The ambient temperature for this installation is maintained at 74 ± 2°F. Therefore, use of 106 ms includes some conservatism and provides an additional margin of 44 ms, over the 150 ms design allocation.

4. When the spurious trips became a problem, the licensee decided to use part of the "dedicated margin" (153 ms) in the form of an intentional time delay. The output relay was removed and an Agastat 0.1-3 minute time delay relay was installed. (The removal of the output relay represents a response time savings of 5-11 ms). The time delay relays were set for a 0.1 sec. delay, with a design allocation of 150 ms maximum. Testing showed that the values were 85-139 ms. While this repeatability is not trivial, it remains within the 150 ms allocation.

5. The 620 ms value used for the transient analysis does not directly equate to the response time requirements in the TSs because the end-points are different. The 620 ms includes 60 ms for release of the roller nuts from the lead screws of the CRD mechanisms. The TS definition of end-point is the opening of the control rod breakers. The original 620 ms value less 60 ms is the 560 ms value which is the value proposed for the TSs. (A similar 60 ms reduction in the allowed response times for other instrumentation shown in TS Table 3.3-2 is also proposed. Our understanding is that all instrumentation meet these revised values.)

The licensee has stated that the start-point for rod drop time measurements is the opening of the CRD breakers. Therefore, the time for the roller nuts to release the control rod before rod motion actually starts is in fact included in the rod drop test.

6. If the 560 ms response time value is further reduced by the original design allocation of 90 ms for "dedicated margin" for uncertainty, we arrive at the present TS value of 470 ms. The appropriateness of such an approach is questionable. The licensee has stated that proposing a TS limit of 0.470 seconds was an error on their part.

We have completed a best-estimate of the present response time of the RCPPM instrumentation. The licensee has conducted certain limited response time measurements on the installed equipment. Measurements have been made from the output of the watt transducer to the output of the contact monitor of the RPS, yielding a worst-case value of 136 ms. This measurement included the bistable and the original output relay. Additionally, this measurement used the output of the RPS contact monitor as its end-point, which is a slight overlap of the RPS measurements. Therefore, we can reasonably subtract 6 ms for the output relay that has been removed, and another 6 ms for the overlap, to yield a value of 124 ms. The newly-installed time delay relays have been measured separately, yielding a worst case value of 139 ms.

The RPS and CRD breaker together have been measured, yielding a worst case value of 104 ms. Recently obtained test data indicates that a value for the watt transducer itself (not including the bistable and output relay) is 22 ms. Based upon manufacturer's calculations, B&W is recommending that the sensor elements be assigned a response time of 16 ms. Collecting these values:

| | | |
|---|---|---|
| a. | sensor elements | 16 ms |
| b. | watt transducer | 22 ms |
| c. | RCPPM measurement | 124 ms |
| d. | time delay measurements | 139 ms |
| e. | RPS measurements | 104 ms |
| | | 405 ms |

The 405 ms best-estimate value is compared to the design limit of 560 ms, and indicates a substantial actual margin of over 150 ms. Further, based upon engineering judgment, we do not believe that even ultra-conservative estimates for uncertainty and for drift over the next 12-18 months could cause the actual response to exceed the 560 ms limit.

Additionally, the licensee has agreed that, at the next plant shutdown of significant duration (i.e., 4 weeks or longer) that occurs after the development of appropriate test methods and procedures (expected to be completed in June 1982), complete response time measurements of at least one RCPPM channel will be made, that include the sensor elements and run to the opening of the CRD breakers. The licensee has stated that the capability already exists to measure the response time of the channel from the output of the sensors. The licensee is concerned that in-place testing of the sensor elements may not be feasible. If this is the case, they have further committed to provide, for our review and approval, an acceptable alternative. If these actual measurements yield results no longer than 560 ms, we could conclude that the system is acceptable.

While this best-estimate of RCPPM response time (being based upon limited testing and original manufacturer's data) may be sufficient for justifying return to full power and continued operation until the next refueling, it is not a sufficient basis for operation over the remaining plant lifetime, 30 years or more.

The licensee has agreed to periodic testing of the response time of one complete RCPPM channel (i.e., including the sensor elements) per pump and RPS at each refueling outage to demonstrate that the required 560 ms limit is maintained. For purposes of this evaluation, the TS limit is:

| | |
|---|---|
| RCPPM channels (including RPS to CRD breakers open) | $\leq$ 0.560 sec. |

Anticipating that the actual measurements will be performed in parts, we are basing the 560 ms limit on:

| | |
|---|---|
| sensor elements | $\leq$ 0.030 sec. |
| RCPPM channel (exclusive of sensor elements, but to RPS contact monitor operating) | $\leq$ 0.325 sec. |
| RPS (including CRD breaker opening) | $\leq$ 0.211 sec. |
| total (includes 6 ms overlap) | $\leq$ 0.566 sec. |

We are assigning a conservative response time of 30 ms to the sensor elements. Such conservatism is necessary since: no response time testing has been conducted, the manufacturer's calculations show quite a variability (3.7 - 14.8 ms), the current being measured in this application is that of a highly inductive load, and the manufacturer has not certified the response time of the sensors. The licensee has agreed to obtain appropriate type test data for the sensor elements within six months. When this data is provided, we can reconsider the present 30 ms assignment.

The question has been raised of using the "staggered test basis" for response time testing of the RCPPM instrumentation. The question is compounded by misleading entries for the RCPPMs (Item 14) in TS Table 3.3-1. Overall, the function of the RCPPM instrumentation is to provide a trip signal upon loss of power to multiple pumps. Therefore, the power to each RCP must be monitored. If each RCP had a single PM channel, the safety function could be accomplished, but the degree of redundancy would be zero. If each RCP had two monitoring channels, the degree of redundancy would be 1; in TS language, the number of redundant channels, N, would be 2. At this plant, since each RCP has two RCPPM channels, N is 2. Therefore, at least one of the two redundant channels for each of the four RCPs must be tested for response time at each refueling outage(i.e., every 18 months). Accordingly, each channel will be retested on intervals of 3 years. Further, considering the equipment involved for this particular instrumentation, a 3-year test interval is the maximum allowable at this time.

In order to avoid future confusion, the licensee has agreed to the following clarification to TS Table 3.3-1 regarding the RCPPMs (Item 14):

| | | |
|---|---|---|
| a. | The Total Number of Channels | 2 per pump |
| b. | Channels to Trip | 1 from 2 or more pumps |
| c. | Min. Channels Operable | 2 per pump |

d. Action Statement 3 is deleted from Item 14.
e. A New Action Statement 25 is added for Item 14, which reads as follows:

"With the number of channels operable one less than the required Minimum Channels Operable requirement, plant operation may continue until the next required Channel Functional Test provided the inoperable channel is placed in the tripped condition within 4 hours."

Additionally, we are concerned that drift in the time delay relays may accumulate between tests. To allay this concern, the licensee has agreed to test all the time delay relays at each refueling not only to assure that their "as found" responses are within the 150 ms allocation, but also to readjust each relay as close as practical back to the nominal 100 ms value.

Based upon the considerations described above, we conclude that the basis of the current plant transient analysis will remain valid for the longer term.

The licensee believes that the cause of the spurious tripping of the RCPPM instrumentation is an electrical power transient occurring on the 6.9 kv station distribution system, causing the sensor elements to trip the channels. The 6.9 kv is obtained from the station main generator via the auxiliary transformer. Considering the RCPPM setpoints of 70% power and the time responses involved, we find it difficult to understand that the main generator is producing such electrical transients. The licensee thinks the transient originates on the electric grid and is being fed back into the station distribution system.

The licensee stated that the non-self-powering elements of the RCPPM channels are powered from the Class 1E Vital Buses. This information does not agree well with the NRC Safety Evaluation (SE) (Reference 1) which originally approved the RCPPMs. The concern in the SE was that certain postulated failure mechanisms (for example, a seismic event) might cause the RCPPMs not to produce an output trip signal. While there may have been some misunderstanding on the original design, the licensee agrees that these postulated failure mechanisms must be addressed. It was therefore agreed that the existing high-level RCP power trip would be required for safety purposes. The licensee has proposed a revised page in the Basis of the TSs to make the reason for this requirement explicit. We believe that having "required" high-level and low-level trips as part of the RCPPMs provides adequate protection to assure that an output trip signal will not be prevented when needed.

We do not believe that the licensee has to date adequately investigated the spurious trips to determine the cause(s). We suggested that voltage recording instruments on the sources of power may be diagnostically necessary. The licensee has agreed to refine the raw data presently available and to further investigate the apparent electric power transients.

Based upon our review of the information provided by the licensee and the reactor vendor, we find that:

(1) The response time of the RCPPMs (through opening of the CRD breakers) is required to be $\leq$ 0.560 seconds for the current plant transient analysis to remain valid.

(2) The addition of an intentional time delay (set to a nominal 100 ms value) is acceptable as long as the 560 ms overall response is maintained.

(3) The best-estimate of the response time for the RCPPM system is adequate for return to full power operation and continued operation to the next scheduled refueling outage. The licensee has agreed to response-time test at least one complete RCPPM channel at the first appropriate outage prior to the refueling outage.

(4) The licensee has committed to response time testing one complete RCPPM system (including sensor elements) for each pump at the next refueling, and each refueling thereafter on a staggered test basis. This will provide adequate assurance that the response time will be acceptable on a continuing basis over the rest of the plant lifetime.

(5) The high-level (pump overpower) trip is required for safety in order to assure that the protective action will occur under certain instrumentation failure conditions.

(6) The licensee is continuing to investigate the spurious tripping to determine the cause(s) of the apparent electrical power transients.

Based upon these findings and upon the actions agreed to by the licensee as discussed in this evaluation, we conclude that the RCPPM instrumentation will provide adequate protection for plant operation at 100% of the authorized 2544 power level.

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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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.

References

1. Letter, J. Stolz (NRC) to J. Hancock (FPC), dated July 21, 1981.
2. Letter, P. Baynard (FPC) to Director, Office of Nuclear Reactor Regulation, NRC, dated November 16, 1981.
3. Letter, J. Stolz (NRC) to J. Hancock (FPC), dated December 4, 1981.
4. Letter, D. Rainey (Babcock & Wilcox) to D. O'Shea (FPC), dated July 17, 1981.
5. Letter, D. Mardis (FPC) to H. Denton (NRC), dated March 4, 1982.
6. Letter, K. Wazowicz (General Electric) to S. Ulm (FPC), dated March 5, 1982.
7. Letter, J. Castanes (Babcock & Wilcox) to P. Baynard (FPC), dated March 5, 1982.
8. Letter, D. Mardis (FPC) to H. Denton (NRC), dated March 9, 1982.

Dated: July 15, 1982

The following NRC personnel have contributed to this Safety Evaluation: J. T. Beard, Gary Holahan and Sydney Miner.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-302

FLORIDA POWER CORPORATION, ET AL

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 55 to Facility Operating License No. DPR-72, issued to the Florida Power Corporation, City of Alachua, City of Bushnell, City of Gainesville, City of Kissimmee, City of Leesburg, City of New Smyrna Beach and Utilities Commission, City of New Smyrna Beach, City of Ocala, Orlando Utilities Commission and City of Orlando, Sebring Utilities Commission, Seminole Electric Cooperative, Inc., and the City of Tallahassee (the licensees) which revised the Technical Specifications (TSs) for operation for the Crystal River Unit No. 3 Nuclear Generating Plant (the facility) located in Citrus County, Florida. Portions of the amendment were authorized by telephone on March 3, 1982, March 9, 1982, April 1, 1982 and April 6, 1982. The administrative addition to the amendment is effective on the date of issuance.

The amendment (1) revises the response time of the Reactor Coolant Pump Power Monitors (RCPPMs), (2) allows operation of the facility at a power level no greater than 2300 MWt (90.4% of full power) with the RCPPMs bypassed and (3) administratively adds limiting conditions for operation and surveillance requirements for the power operated relief valves which had inadvertently been omitted from a previous amendment. Portions of the amendment were authorized on an expedited basis to allow plant startup and operation at steady state while problems associated with the RCPPMs are being resolved.

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The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. 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 March 4, 1982, and April 1, 1982, as supplemented by letters dated March 9, 1982, April 6, 1982, and April 30, 1982, (2) the Commission's letters to Florida Power Corporation dated March 12, 1982, April 2, 1982, and April 16, 1982, (3) Amendment No. 55 to License No. DPR-72, and (4) 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, NW, Washington, D.C., and at the Crystal River Public Library, 668 N.W. First Avenue, Crystal River, Florida. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 15th day of July 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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