April 3, 1978

Docket No.: 50-302

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JRBuchanan
RBaer

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. 14 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in partial response to your applications dated November 8, 1977, and December 14, 1977, as modified by you following discussions with the staff.

This amendment modifies the Technical Specifications to revise reactor internals vent valve surveillance requirements and to indicate a plant staff reorganization. The remaining portions of your November 8 and December 14 applications are being treated separately.

As discussed in the enclosed Safety Evaluation, you are requested to submit, within 90 days from the date of this letter, a change to the Quality Assurance program description for CR-3 to be consistent with the changes made to the plant staff organization.

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

Sincerely,

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

Enclosures:
1. Amendment No. 14
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

*SEE PREVIOUS YELLOW FOR CONCURRENCES

Const. 1

GD

OFFICE	ORB#4:DOR	ORB#4:DOR	STSG:DOR	C-RSB-OT:DOR	OELD	C-ORB#4:DOR
SURNAME	RIngram *	CNelson:rm	JMcGough *	RBaer *	SHLewis *	RReid
DATE	3/ /78	4/ /78	3/ /78	3/ /78	3/ /78	4/3/78

NRC FORM 318 (9-76) NRCM 0240 ☆ U.S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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

Sincerely,

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

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

cc w/enclosures: See next page

CHB for

Concur subject to changes made in SE & letter & discussed with C Nelson

OFFICE	ORB#4:DOR	ORB#4:DOR	STSG:DOR	C-RSB-OT:DOR	OELD SHL	C-ORB#4:DOR
SURNAME	RIngram	~~XX~~ CNelson:rm	JMcGough	RBaer	SHLewis	RReid
DATE	3/21/78	3/21/78	3/1/78	3/29/78	3/31/78	3/ /78

NRC FORM 318 (9-76) NRCM 0240

Florida Power Corporation

cc w/enclosures:

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.: 12/14/77
Bureau of Intergovernmental Relations
660 Apalchee Parkway
Tallahassee, Florida 32304

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

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

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
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 November 8 and December 14, 1977, 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. 14, 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: April 3, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 14

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

3/4 4-32
B 3/4 4-13
6-3
6-5

3.4.10 STRUCTURAL INTEGRITY

ASME CODE CLASS 1, 2 and 3 COMPONENTS

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.10.1.

APPLICABILITY: All MODES.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the component(s) to within its limit or isolate the affected component(s) from service.

d. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1 In addition to the requirements of Specification 4.0.5:

a. The reactor coolant pump flywheels shall be inspected per the recommendations of Regulatory Position C.4.b. of Regulatory Guide 1.14, Revision 1, August 1975.

SURVEILLANCE REQUIREMENTS (Continued)

b. Each internals vent valve shall be demonstrated OPERABLE at least once per 18 months* during shutdown, by:

1. Verifying through visual inspection that the valve body and valve disc exhibit no abnormal degradation,

2. Verifying the valve is not stuck in an open position, and

3. Verifying through manual actuation that the valve is fully open with a force of $\leq$ 425 lbs (applied vertically upward).

*The first periodic surveillance of the internals vent valves may be performed up to 24 months following the original surveillance (plus 25% per 4.0.2) but no later than the first refueling outage.

BASES

All pressure-temperature limit curves are applicable up to the fifth effective full power year. The protection against non-ductile failure is assured by maintaining the coolant pressure below the upper limits of Figures 3.4-2, and 3.4-3, and 3.4-4.

The pressure and temperature limits shown on Figures 3.4-2 and 3.4-3 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50.

The number of reactor vessel irradiation surveillance specimens and the frequencies for removing and testing these specimens are provided in Table 4.4-3 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

The limitations imposed on pressurizer heatup and cooldown and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

3/4.4.10 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2 and 3 components, except steam generator tubes, ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant. To the extent applicable, the inspection program for these components is in compliance with Section XI of the ASME Boiler and Pressure Vessel Code.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves 1) ensure OPERABILITY, 2) ensure that the valves are not stuck open during normal operation, and 3) demonstrate that the valves are fully open at the forces assumed in the safety analysis.

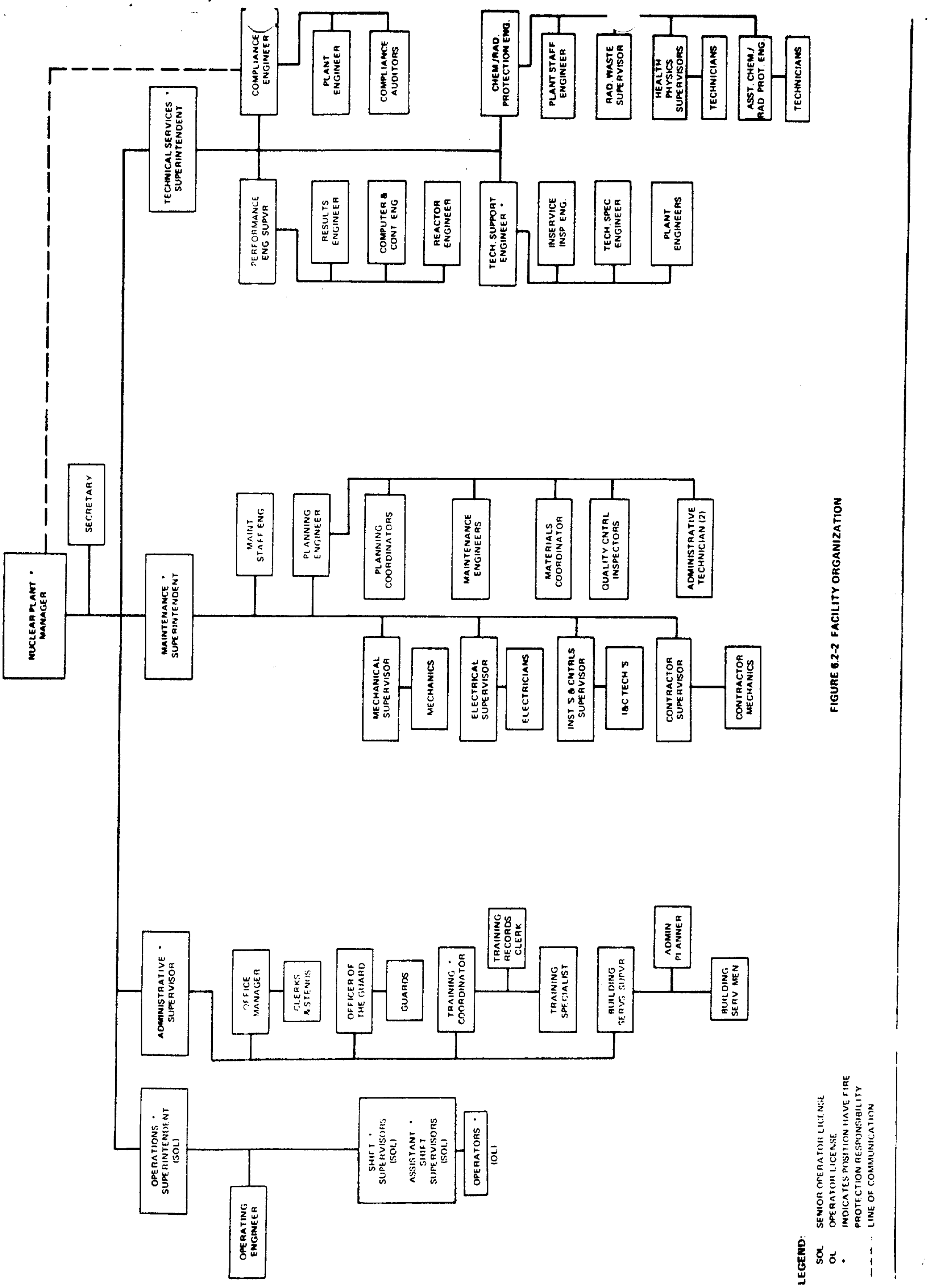

FIGURE 6.2-2 FACILITY ORGANIZATION

TABLE 6.2-1

MINIMUM SHIFT CREW COMPOSITION#

LICENSE CATEGORY	APPLICABLE MODES	
	1, 2, 3 & 4	5 & 6
SOL	1	1*
OL	2	1
Non-Licensed	3	1

*Does not include the licensed Senior Reactor Operator or Senior Reactor Operator Limited to Fuel Handling Individual supervising CORE ALTERATIONS after the initial fuel loading.

#Shift crew composition may be less that the minumum requirement for a period of time not to exceed 2 hours in order to accomodate unexpected absence of on duty shift crew members provided immediate action is taken to restore the shift crew composition to within the minimum requirements of Table 6.2-1.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Chemistry and Radiation Protection Engineer who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Nuclear Plant Manager and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A training program for the Fire Brigade shall be maintained under the direction of the Nuclear Plant Manager and shall meet or exceed the requirements of Section 27 of the NFPA Code-1976, except for Fire Brigade training sessions which shall be held at least quarterly.

6.5 REVIEW AND AUDIT

6.5.1 PLANT REVIEW COMMITTEE (PRC)

FUNCTION

6.5.1.1 The Plant Review Committee shall function to advise the Nuclear Plant Manager on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The Plant Review Committee shall be composed of the:

Chairman:	Technical Services Superintendent
Member:	Operations Superintendent
Member:	Technical Support Engineer
Member:	Maintenance Superintendent
Member:	Chemistry and Radiation Protection Engineer
Member:	At large (Designated by Chairman)

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the PRC Chairman to serve on a temporary basis; no more than two alternates shall participate as voting members in PRC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The PRC shall meet at least once per calendar month and as convened by the PRC Chariman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the PRC shall consist of the Chariman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The Plant Review Committee shall be responsbile for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Nuclear Plant Manager to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear Safety.

c. Review of all proposed changes to the Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Director-Power Production and to the Chairman of the Nuclear General Review Committee.

f. Review of events requiring 24 hour written notification to the Commission.

g. Review of facility operations to detect potential nuclear safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Nuclear General Review Committee.

i. Review of the Plant Security Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear General Review Committee.

j. Review of the Emergency Plan and implementing procedures and shall submit recommended changes to the Chairman of the Nuclear General Review Committee.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 14 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 November 8, 1977, and December 14, 1977, Florida Power Corporation (FPC) proposed changes to the Crystal River Unit No. 3 (CR-3) Technical Specifications. The proposed changes would (1) revise reactor internals vent valve surveillance requirements and (2) indicate a plant staff reorganization.

Evaluation

1. By letter dated November 8, 1977, FPC proposed a change to the surveillance testing of reactor internals vent valves to better define the force needed to fully open the internals vent valves and to delete the requirement to measure the opening force of the valves. The present Technical Specification requires manual actuation of the vent valves to verify that the valve begins to open from the fully closed position with a force equivalent to $\leq .15$ psid and is fully open with a force equivalent to $\leq .30$ psid. The proposed change would require the vent valves to be open at the force equivalent to 1.0 psid. FPC has shown that this change increases the calculated peak cladding temperature (PCT) during the limiting Loss of Coolant Accident (LOCA) by less than 3°F. This is not considered a significant increase and does not cause the limiting LOCA PCT to exceed any of the 10 CFR 50.46 criteria, nor does this change affect which LOCA break is limiting. FPC's analyses, presented as justification for this proposed change, have been reviewed by the staff and found to be acceptable. Based on this information and the continued surveillance requirements on the reactor internals vent valves, we find this change to be acceptable.

FPC also proposed to extend the surveillance interval for these vent valves from 18 to 24 months. This would allow this surveillance to be performed at the first refueling outage currently scheduled to begin no earlier than November 1978. Since the head must be removed from the reactor vessel for this surveillance, it is the intent of the Technical Specifications to require that the surveillance be performed during refueling outages which normally, following the first, are less than 18 months apart. We, therefore, agree that a 24-month surveillance interval (plus the 25% allowed by Technical Specification 4.0.2) is appropriate to allow performance during the first refueling outage. However, the 18 month interval in the Technical Specifications should remain applicable to subsequent refueling outages. After discussions with the staff, FPC has, therefore, agreed that it will be sufficient to add a footnote to Technical Specification 4.4.10.1.b allowing the first surveillance interval to be 24 months.

2.

management level between the Compliance Engineer and the Plant Manager and between the Chemical/Radiation Protection Engineer and the Plant Manager. This change is not contrary to established NRC policy. However, the statement made by FPC in their submittal, that the Compliance Engineer has direct access to the Nuclear Plant Manager on "any problems", should be shown on Figure 6.2-2 by a dotted line (plus a footnote describing its meaning) from the Compliance Engineer to the Plant Manager.

This has been discussed with FPC and they have agreed to this modification. In addition, FPC indicated that they do not consider the compliance function a formal part of the Quality Assurance (QA) program. The Compliance Engineer does have Quality Assurance/Quality Control functions as described in Section 1.7.6.7.2 of the current QA program. This has been discussed with FPC and they agree that the Compliance Engineer does have a formal role in the existing QA Program.

The licensee has also proposed that the Technical Specifications be altered to reflect changed position titles. We have reviewed these changes and conclude that the plant staff organization arrangement maintains acceptable lines of authority, includes the necessary areas of expertise, and is acceptable.

The list of members of the Plant Review Committee (PRC) would also be changed to reflect the proposed changes in the plant staff organization. We find that the PRC will continue to be composed of persons representing the various disciplines required for effective review of plant operations. This change is also consistent with the Standard Technical Specifications in this area. We have concluded, therefore, that this change is acceptable.

FPC has been requested to amend Section 1.7 of the Final Safety Analysis Report to modify the QA program description to be consistent with the changes made to the plant staff organization.

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: April 3, 1978

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. 14 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 Technical Specifications for operation of the Crystal River Unit No. 3 Nuclear Generating Plant located in Citrus County, Florida. The amendment is effective as of the date of issuance.

This amendment modifies the Technical Specifications to revise reactor internals vent valve surveillance requirements and to indicate a plant staff reorganization.

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 November 8 and December 14, 1977, (2) Amendment No. 14 to License No. DPR-72, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Crystal River Public Library, Crystal River, Florida. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 3rd day of April 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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