July 24, 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. 15 to Facility Operating License No. DPR-72 for Crystal River Unit No. 3 Nuclear Generating Plant. This amendment consists of changes to the License and its appended Technical Specifications in response to your application dated June 28, 1978, and in partial response to your application dated May 30, 1978. As discussed with your staff, we will evaluate your proposed change No. 26, part of your May 30, 1978 submittal, in conjunction with our review of the CR-3 fire protection program.

This amendment authorizes you to receive and possess at CR-3 four spent fuel assemblies from Oconee-1 which you intend to use as fuel in the CR-3 reactor during the remainder of Cycle 1. Our review of the use of these assemblies as fuel at CR-3 will commence upon our receipt of the appropriate submittal. This amendment also revises Technical Specifications to reflect a change in the reactor vessel surveillance capsule installation and removal schedule.

Copies of the Safety Evaluation/Environmental Impact Appraisal and Notice of Issuance/Negative Declaration are also enclosed.

Sincerely,

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

Enclosures and cc:
See next page

7/21/78

Const. 1
GD

OFFICE➤	ORB#4:DOR	C-EEB:DOR	NRR/AG	OELD	STSG:DOR	C-ORB#4
SURNAME➤	CNelson:dn	GKNighton		SHLewis	JMcGough	RRe
DATE➤	7/21/78	7/21/78	7/21/78	7/24/78	7/24/78	7/

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Enclosures:
1. Amendment No. 15
2. Safety Evaluation/Environmental
   Impact Appraisal
3. Notice/Negative Declaration

cc w/enclosures: See next page

OFFICE➤						
SURNAME➤						
DATE➤						

Florida Power Corporation

cc w/enclosure(s):
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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
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345 Courtland Street, N.E.
Atlanta, Georgia 30308

Chief, Energy Systems Analyses
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Office of Radiation Programs
U. S. Environmental Protection Agency
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Crystal River Public Library
Crystal River, Florida 32629

cc w/enclosures and incoming
dtd: 5/30 & 6/28/78
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. 15
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 May 30 and June 28, 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 application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, Facility Operating License No. DPR-72 is hereby amended as indicated below and by changes to the Technical Specifications as indicated in the attachment to this license amendment:

   Add a new paragraph 2.B.(7) to read as follows:

   2.B.(7) Florida Power Company, pursuant to the Act and 10 CFR Parts 30 and 70, to receive and possess, but not separate, that by-product and special nuclear materials associated with four (4) fuel assemblies (B&W Identification Numbers 1A-01, 04, 05 and 36 which were previously irradiated in the Oconee Nuclear Station, Unit No. 1) acquired by Florida Power Corporation from Duke Power Company for use as reactor fuel in the facility.

   Revise Paragraph 2.C.(2) to read as follows:

   2.C.(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 15 , 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: July 24, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 15

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 4-29
B 3/4 4-12
B 3/4 4-13

recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure.

The closure head region is significantly stressed at relatively low temperatures (due to mechanical loads resulting from bolt pre-load). This region largely controls the pressure-temperature limitations of the first several service periods. The outlet nozzles of the reactor vessel also affect the pressure-temperature limit curves of the first several service periods. This is due to the high local stresses at the inside corner of the nozzle which can be two to three times the membrane stresses of the shell. After the first several years of neutron radiation exposure, the RT_{NDT} temperature of the beltline region materials will be high enough so that the beltline region of the reactor vessel will start to control the pressure-temperature limitations of the reactor coolant pressure boundary. For the service period for which the limit curves are established, the maximum allowable pressure as a function of fluid temperature is obtained through a point-by-point comparison of the limits imposed by the closure head region, outlet nozzles, and beltline region. The maximum allowable pressure is taken to be the lower pressure of the three calculated pressures. The calculated pressure temperature limit curves are then adjusted by 25 psi and 10°F for possible errors in the pressure and temperature sensing instruments. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all operating reactor coolant pump combinations. The limit curves were prepared based upon the most limiting adjusted reference temperature of all the beltline region materials at the end of the fifth effective full power year. The fifth effective full power year was selected because the second surveillance capsule will be withdrawn at the end of the fifth cycle. The time difference between the fifth cycle and fifth effective full power year provides adequate time for establishing the operating pressure and temperature limitations for the period of operation after the fifth effective full power year.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10 CFR 50, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside the radius are essentially identical, the measured transition shift for a sample can be applied

BASES

with confidence to the adjacent section of the reactor vessel. The limit curves must be recalculated when the ΔRT_{NDT} determined from the surveillance capsule is different from the calculated ΔRT_{NDT} for the equivalent capsule radiation exposure. Since the first surveillance program capsule will be withdrawn at 270 $\pm$ EFPD and the limit curves were prepared based upon the predicted impact properties at the end of the fifth effective full power year, it is predicted that no readjustment will be required to the limit curves for the first 5 effective full power years. Adjustment may be required after the withdrawal of the second capsule.

The unirradiated transverse impact properties of the beltline region materials, required by Appendices G and H to 10 CFR 50, were determined for those materials for which sufficient amounts of material were available. The unirradiated impact properties and residual elements of the beltline region materials are listed in Bases Table 4-1. The adjusted reference temperature are calculated by adding the predicted radiation-induced ΔRT_{NDT} and the unirradiated. The predicted ΔRT_{NDT} are calculated using the respective neutron fluence and copper and phosphorus contents. Bases Figure 4-1 illustrates the calculated peak neutron fluence, at several locations through the reactor vessel beltline region wall and at the center of the surveillance capsules as a function of exposure time.

Bases Figure 4-2 illustrates the design curves for predicting the radiation induced ΔRT_{NDT} as a function of the material's copper and phosphorus content and neutron fluence. The adjusted RT_{NDT}'s of the beltline region materials at the end of the fifth full power year are listed in Bases Table 4-1. The adjusted RT_{NDT}'s are given for the 1/4T and 3/4T (T is wall thickness) vessel wall locations. The assumed RT_{NDT} of the closure head region and of the outlet nozzle steel forgings is 60°F.

During cooldown at the higher temperatures, the limits are imposed by thermal and loading cycles on the steam generator tubes. These limits are the vertical segments of the limit lines on Figures 3.4-3 and 3.4-4, respectively. These limits will not require adjustments due to the neutron fluences.

Figure 3.4-2 presents the pressure-temperature limit curve for normal heatup. This figure also presents the core criticality limits as required by Appendix G to 10 CFR 50. Figure 3.4-3 presents the pressure temperature limit curve for normal cooldown. Figure 3.4-4 presents the pressure-temperature limit curves for heatup and cooldown for inservice leak and hydrostatic testing.

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 3.4-4 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 4.4-5 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.

TABLE 4.4-5

REACTOR VESSEL MATERIAL IRRADIATION SURVEILLANCE SCHEDULE

Capsule	Installation	Removal
A	At 270 $\pm$ 10 EFPD of First Cycle	Standby
C	At 270 $\pm$ 10 EFPD of First Cycle	End of Ninth Cycle
E	At 270 $\pm$ 10 EFPD of First Cycle	Standby
B	Initial Fuel Load	At 270 $\pm$ 10 EFPD of First Cycle
D	Initial Fuel Load	End of Fifth Cycle
F	At 270 $\pm$ 10 EFPD of First Cycle	End of Fifth Cycle

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup and cooldown of 100°F in any one hour period, and

b. A maximum spray water temperature differential of 410°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperature shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit once per 12 hours during auxiliary spray operation with pressurizer temperature > 440°F.

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

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 15 TO LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

Introduction

Crystal River Unit 3 Nuclear Generating Plant (CR-3) is currently shutdown to repair damage caused by the failure of Burnable Poison Rod Assemblies. As part of the repair effort, the reactor was defueled. On June 9, 1978, a plant-fabricated rigging hook on the missile shield crane failed, dropping a test weight on fuel assembly A-48 in the spent fuel pool. Inspection of this assembly revealed that deformation had occurred sufficient to preclude its further use as fuel.

In light of the above, additional fuel must be obtained for CR-3 to restart and complete Cycle 1. By letter of June 28, 1978, Florida Power Corporation (FPC) requested a license amendment which would allow them to obtain four fuel assemblies previously irradiated at Oconee Nuclear Station Unit No. 1 (Oconee-1), for this purpose. These would replace the damaged assembly and its three symmetrical assemblies to minimize quadrant variations. Our evaluation of FPC's possession of the four Oconee-1 assemblies at CR-3 follows. Our evaluation does not address the use of the Oconee assemblies as fuel in the CR-3 reactor. This will be handled as a separate action.

In a separate application, dated May 30, 1978, FPC proposed changes to their Appendix A Technical Specifications to modify the reactor vessel surveillance capsule removal and installation schedule. We have also reviewed this change.

I-Safety Evaluation

A. Possession of Four Oconee-1 Fuel Assemblies

Paragraph 2.B.(6) of the CR-3 license currently reads as follows:

> B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:
>
> (6) Florida Power Corporation, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

FPC has proposed to modify paragraph 2.B.(6) to read:

> (6) Florida Power Corporation, pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such by-product and special nuclear materials as may be produced by the operation of the facility and that by-product and special nuclear materials associated with four (4) fuel assemblies acquired by Florida Power Corporation from Duke Power Company which were previously irradiated in the Oconee Nuclear Station, Unit One.

In their June 28, 1978 submittal FPC states:

> "It is our intent to transport these four (4) assemblies to Crystal River Unit No. 3 and receive them for use in the reactor for the remainder of Cycle 1..."

The safety considerations associated with this change are limited to activities associated with the handling and storage of the four Oconee assemblies at CR-3.

As discussed in section 4 of the NRC staff's Safety Evaluation Report (SER) (issued July 5, 1974) supporting issuance of an operating license for CR-3, the fuel used at CR-3 and Oconee-1 are similar in design. Dimensions of the Oconee-1 and CR-3 fuel assemblies are identical and therefore the Oconee fuel will fit in the spent fuel storage locations at CR-3. In addition the initial U235 enrichment of the Oconee fuel (2.10 w/o) is less than the average enrichment of CR-3 Cycle 1 fuel (2.44 w/o, FSAR Table 3-2). Therefore, we have concluded that the Oconee fuel can be stored safely in the spent fuel storage configuration at CR-3 ($>$ 21" center to center spacing).

Section 9.6 of the CR-3 Final Safety Analysis Report (FSAR) addresses the handling of spent fuel, including an analysis of dropping a 100 ton-ten element shipping cask in the cask storage area and in the spent fuel pool adjacent to the cask storage area. Section 9.1.2 of our July 5, 1974, SER presents our review of the cask drop accidents and our conclusion that the design of the CR-3 spent fuel storage facility and the consequences of a cask drop accident are acceptable. Technical Specifications which require that (1) no fuel be in the pool adjacent to the cask storage area, (2) the watertight gate between storage pools be in place and sealed; and (3) the crane interlocks which prevent cask travel over the storage pools be operable, insure that our previous evaluation and conclusions are still valid.

FPC's submittal references an FSAR analysis of a 100-ton 10 element cask drop from 43 feet in the cask loading area for rail shipment. This analysis concludes that the release of all gap activity from the fuel elements (120 days decay) could result in site boundry doses of 0.23 rem (whole body) and 0.14 rem (thyroid) which are well within 10 CFR Part 100 guidelines. FPC also states that a 25 ton-one element cask will be used to transfer the Oconee fuel and therefore the FSAR analysis bounds the use of this cask.

The above FSAR analysis did not address the potential for loss of fuel element cooling and subsequent fuel melting. Therefore, we performed an independent analysis of the one Oconee element cask drop assuming loss of cooling, 120 days fuel decay, and the fraction of noble gases and iodines released as 100% and 50%, respectively. The results of our analysis are exclusion area boundary doses of 0.5 rem (thyroid) and 5×10^{-5} rem (whole body). These consequences are well within 10 CFR Part 100 guidelines. Since the Oconee fuel has been decaying since the end of the Oconee-1 first cycle (greater than 1300 days), the above analysis is conservative.

Based on our evaluation of the cask drop accidents and the spent fuel storage capabilities at CR-3, we conclude that the storage and handling of the four Oconee fuel elements at CR-3 are acceptable.

We have also reviewed the indemnity considerations related to the location of the Oconee fuel assemblies at CR-3. The licensees for CR-3 currently have in effect with the Commission an Indemnity Agreement (No. B-54) in the form specified in 10 CFR §140.92. Article I, section 9, of this regulation defines the radioactive material subject to the Indemnity Agreement as "source, special nuclear, and byproduct material which (1) is used or to be used in, or is irradiated or to be irradiated by, the nuclear reactor or reactors subject to the license or licenses designated in the Attachment hereto, or (2) which is produced as the result of operation of said reactor(s)."

Since the four fuel assemblies to be transferred to CR-3 are to be used as fuel in the CR-3 reactor, no change to the CR-3 Indemnity Agreement is necessary for this action.

We have revised the wording of the license amendment, as proposed by FPC, to state the authorization for FPC's possession of Oconee-1 fuel in a separate paragraph, to specify, by B&W identification numbers, the Oconee-1 fuel to be possessed and to limit this authorization to possession of the Oconee-1 assemblies for use as fuel in CR-3. These changes have been discussed with and agreed to by the licensee.

B. Surveillance Capsule Removal and Installation Schedule

FPC proposed that CR-3 Capsule B be removed at 270 EFPD in lieu of at the end of the first fuel cycle and that CR-3 Capsules A, C, E and F be installed at 270 EFPD in lieu of at the end of the first fuel cycle. This change would allow FPC to take advantage of the access to the capsules provided by the current outage.

Appendix H of 10 CFR Part 50 requires the first capsule to be withdrawn when the ΔRT_{NDT} reaches $50^{o}F$ or 1/4 of service life, whichever is earlier. For CR-3, a $50^{o}F$ shift in RT_{NDT} has already occurred. Therefore, withdrawal of Capsule B at 270 EFPD will meet the requirements of Appendix H and will provide the necessary information to check the temperature-pressure limit curves of this plant. Also, the installation of capsules A, C, E and F at 270 EFPD will give these capsules approximately 200 EFPD of additional exposure. This additional exposure will provide us with radiation damage data on these capsule materials at slightly higher fluence levels than the original schedule would. This additional exposure will in no way reduce the usefulness of the surveillance data that will be obtained from these CR-3 capsules. Also, this schedule change will not adversely affect the Integrated Reactor Vessel Material Surveillance Program that CR-3 is committed to.

Based on the above, we conclude that the proposed change in the capsule removal and installation schedule is acceptable.

Conclusion on Safety

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.

II. Environmental Impact Appraisal Regarding Transfer of Oconee Fuel Assemblies to CR-3

We have evaluated the potential environmental impact associated with the license amendment proposed on June 28, 1978, as required by the National Environmental Policy Act and Section 51.7 of 10 CFR Part 51.

There will be four separate shipments by truck of a single irradiated fuel assembly from Oconee to CR-3. Each of the four fuel assemblies has decayed at least 1300 days. The distance each shipment will travel between Oconee and CR-3 is about 480 miles. The thermal power level per fuel assembly for Oconee is about the same for CR-3.

Shipment of spent fuel from CR-3 to the reprocessing facility in Barnwell, South Carolina, was considered in the Final Environmental Statement (FES) dated May 1973. We estimated 10 shipments per year to transport the irradiated fuel from CR-3 with six fuel assemblies per cask and one cask per shipment. The shipments were to be made by rail, a distance of about 350 miles. The irradiated fuel would be shipped after a 120 to 150-day cooling period.

We also estimated in the FES that there might be cumulative dose of 0.16 man rem, during each rail shipment, to the general public along the route and to the workers transporting the spent fuel. We have reviewed the basis for this estimate and conclude it is a conservative estimate of the man rem exposure for a shipment of a single fuel assembly from Oconee to CR-3. Therefore, we estimate that the radiation exposure during the four shipments from Oconee to CR-3 should be less than 0.7 man rem to the general population and workers transporting the fuel. This is a small fraction of the fluctuations in the annual dose this population would receive from natural background radiation.

We have also estimated the exposure to the workers removing the spent fuel from the Oconee spent fuel pool and placing the spent fuel in the CR-3 pool. This exposure should be less than one man rem. This is based on relevant assumptions for occupancy times and dose rates in the spent fuel pool area from radionuclide concentrations in the water. This additional exposure is less than 0.2% of the total annual occupational radiation burden at either facility.

Based on the above, we conclude that the shipment of four assemblies from Oconee to CR-3 will not result in any significant increase in doses received by the public or by occupational workers.

The four shipments of spent fuel from Oconee to CR-3 are estimated to be 1% of the total number of shipments of spent fuel from CR-3, during its 40-year lifetime, considered in the FES. This small increase in the number of shipments of spent fuel associated with the operation of CR-3 will not change the conclusions of the FES and will not have any significant environmental impact.

Conclusion and Basis for Negative Declaration

On the basis of our evaluation and information supplied by the licensees, it is concluded that the implementation of this proposed change will have no significant impact on the environment other than that already predicted and described in our FES and subsequent environmental impact appraisals.

Having reached these conclusions, the Commission has determined that an environmental impact statement need not be prepared for this proposed change and that a Negative Declaration to that effect should be issued.

III. Environmental Conclusion Regarding Surveillance Capsule Schedule Change

We have determined that this change 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 this change 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 change.

Dated: July 24, 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

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 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 license and its appended 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: 1) authorizes Florida Power Corporation (FPC) to receive and possess at the facility four spent fuel assemblies from Oconee Nuclear Station, Unit No. 1, which FPC is expected to request

Commission authorization to use as fuel in the facility's reactor during the remainder of Cycle 1 operation; and 2) revises the Technical Specifications to reflect a change in the reactor vessel surveillance capsule installation and removal schedule.

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 prepared an environmental impact appraisal for Item 1, above, of this amendment and has concluded that an environmental impact statement for this particular action is not warranted because there will be no significant environmental impact attributable to the action other than that which has already been predicted and described in the Commission's Final Environmental Statement for the facility dated May 1973.

The Commission has determined that the issuance of Item 2, above, of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an 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 May 30, 1978 (Proposed Change No. 28) and June 28, 1978, (2) Amendment No. 15 to License No. DPR-72, and (3) the Commission's related Safety Evaluation/Environmental Impact Appraisal. 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 24th day of July 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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