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Docket No. 50-302



Mr. J. A. Hancock
Director, Nuclear Operations
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Hancock:

The Commission has issued the enclosed Amendment No. 36 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant in response to your application dated March 17, 1978, as supplemented.

This amendment revises the Appendix A Technical Specifications to authorize an increase in the capacity of the spent fuel storage pools at Crystal River Unit No. 3. Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff. We have also corrected page 5-4 of the Technical Specifications to correct the amount of uranium per fuel rod to that which was reviewed and accepted in our August 1, 1980 review of Cycle 3.

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

Sincerely,

Original signed by
Robert W. Reid

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

Enclosures:
1. Amendment No. 36 to DPR-72
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice/Negative Declaration

cc w/enclosures:
See next page

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AMENDMENT

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555
November 18, 1980

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Docket No. 50-302

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: CRYSTAL RIVER UNIT NO. 3

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies ( 12 ) of the Notice are enclosed for your use.

☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).

☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.

☐ Notice of Availability of Applicant's Environmental Report.

☐ Notice of Proposed Issuance of Amendment to Facility Operating License.

☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.

☐ Notice of Availability of NRC Draft/Final Environmental Statement.

☐ Notice of Limited Work Authorization.

☐ Notice of Availability of Safety Evaluation Report.

☐ Notice of Issuance of Construction Permit(s).

☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).

☒ Other: Amendment No. 36
Referenced documents have been provided PDR.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

| | | | | | | |
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NRC FORM 102 (1-76)

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

November 17, 1980

Docket No. 50-302

Mr. J. A. Hancock
Director, Nuclear Operations
Florida Power Corporation
P. O. Box 14042, Mail Stop C-4
St. Petersburg, Florida 33733

Dear Mr. Hancock:

The Commission has issued the enclosed Amendment No. 36 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant in response to your application dated March 17, 1978, as supplemented.

This amendment revises the Appendix A Technical Specifications to authorize an increase in the capacity of the spent fuel storage pools at Crystal River Unit No. 3. Some portions of your proposed Technical Specifications have been modified to meet our requirements. These modifications have been discussed with and agreed to by your staff. We have also corrected page 5-4 of the Technical Specifications to correct the amount of uranium per fuel rod to that which was reviewed and accepted in our August 1, 1980 review of Cycle 3.

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

Sincerely,

Robert W. Reid

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

Enclosures:
1. Amendment No. 36 to DPR-72
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice/Negative Declaration

cc w/enclosures:
See next page

cc w/enclosure(s):

Mr. S. A. Brandimore
Vice President and General Counsel
P. O. Box 14042
St. Petersburg, Florida 33733

Mr. Wilbur Langely, Chairman
Board of County Commissioners
Citrus County
Iverness, Florida 36250

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

Director, Criteria and Standards
Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

Crystal River Public Library
Crystal River, Florida 32629

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Power Plant Siting Section
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Tallahassee, Florida 32301

Attorney General
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Los Alamos, New Mexico 87544

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Nuclear Power Generation Division
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Bethesda, Maryland 20014

Mr. Tom Stetka, Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 2110
Crystal River, Florida 32629

cc w/enclosure(s) & incoming dtd.:
1/9, 3/3, 3/17, 3/22 & 8/30/78, 1/18,
3/16, 6/29, 9/5, 10/1, 10/10 & 12/5/79

Bureau of Intergovernmental Relations
660 Apalachee Parkway
Tallahassee, Florida 32304

Ms. Lori Spence
Westinghouse Electric Corporation
Monroeville Nuclear Center (Bay 413)
Box 355
Pittsburgh, Pennsylvania 15230

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. 36
License No. DPR-72

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

   A. The application for amendment by Florida Power Corporation, et al (the licensees) dated March 17, 1978, as supplemented, complies 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.

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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) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 36, 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 Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 17, 1980

ATTACHMENT TO LICENSE AMENDMENT NO. 36

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of 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.

3/4 9-7

5-4

5-5

5-6

## CRANE TRAVEL - SPENT FUEL STORAGE POOL BUILDING

### LIMITING CONDITION FOR OPERATION

3.9.7 Loads in excess of 2750 pounds, except for movement of the missile shield and pool divider gate as necessary for access to the fuel assemblies, shall be prohibited from travel over fuel assemblies in the storage pool.*

APPLICABILITY: With fuel assemblies and water in the storage pool.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.7.1 Crane interlocks and/or physical stops which prevent crane travel with loads in excess of 2750 pounds over fuel assemblies shall be demonstrated OPERABLE within 7 days prior to crane operation and at least once per 7 days during crane operation.

4.9.7.2 Prior to operating the crane in the cask handling mode, verify that:

a. No fuel assemblies are in the storage pool adjacent to the cask loading area, and

b. The watertight gate between storage pools is in place and sealed.

*except for the removal of old spent fuel racks and the installation of the high density spent fuel storage racks in the spent fuel storage pool. The missile shield shall cover the spent fuel in the alternate pool during rack handling.

## COOLANT CIRCULATION

### LIMITING CONDITION FOR OPERATION

3.9.8 At least one decay heat removal loop shall be in operation.

APPLICABILITY: MODE 6.

ACTION:

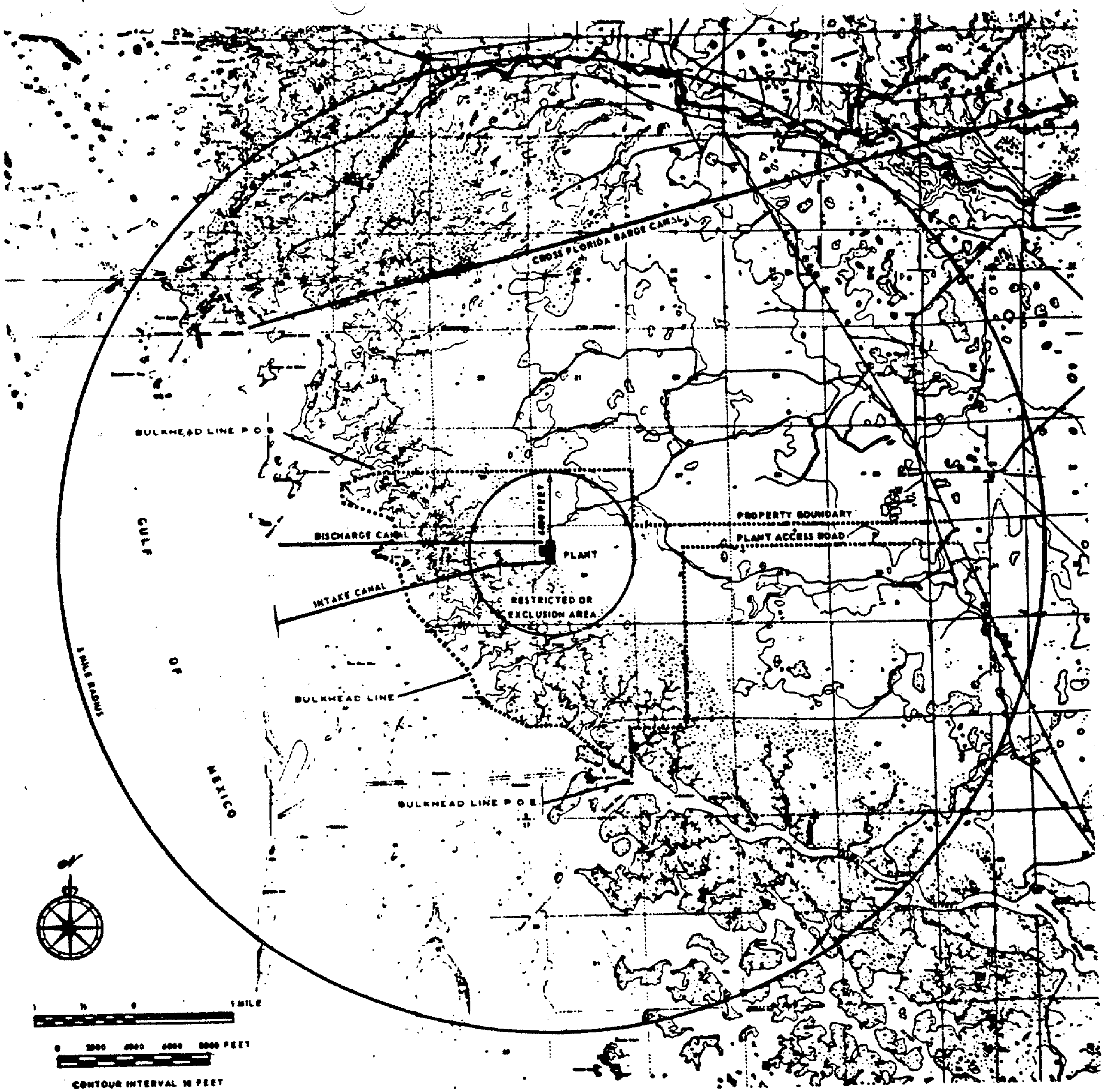
a. With less than one decay heat removal loop in operation, except as provided in b. below, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the Reactor Coolant System. Close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere within 4 hours.

b. The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of CORE ALTERATIONS to prevent water turbulence problems.

c. The provisions of Specification 3.0.3 are not applicable.

### SURVEILLANCE REQUIREMENTS

4.9.8 A decay heat removal loop shall be determined to be operating and circulating reactor coolant at a flow rate of $\geq$ 2700 gpm at least once per 24 hours.




LOW POPULATION ZONE

FIGURE 5.1-2

## DESIGN PRESSURE AND TEMPERATURE

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

## 5.3 REACTOR CORE

### FUEL ASSEMBLIES

5.3.1 The rector core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2253 grams uranium. The initial core loading shall have a maximum enrichment of 2.83 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.30 weight percent U-235.

### CONTROL RODS

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

## 5.4 REACTOR COOLANT SYSTEM

### DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 4.1.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670°F.

### VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,180 ± 200 cubic feet at a nominal $T_{avg}$ of 525°F.

## 5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

## 5.6 FUEL STORAGE

### CRITICALITY

5.6.1 The new fuel storage racks are designed and shall be maintained with a nominal 21-1/8 inch center-to-center distance between fuel assemblies placed in the storage racks. The high density spent fuel storage racks are designed and shall be maintained with a nominal 10.5 inch center-to-center distance between fuel assemblies placed in the storage racks. Both of these rack designs ensure a $k_{eff}$ equivalent to $\leq 0.95$ with the storage pool filled with unborated water. The $k_{eff}$ of $\leq 0.95$ includes a conservative allowance of $> 1\% \Delta k/k$ for uncertainties. In addition, fuel in the new and spent fuel storage racks shall have a U-235 loading of $\leq 42.7$ grams of U-235 per axial centimeter of fuel assembly ($\leq$an enrichment of 3.3 3.3 weight percent U-235).

### DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 138 feet 4 inches.

DESIGN FEATURES

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1153 fuel assemblies and 6 failed fuel containers.

## 5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limit of Table 5.7-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 36 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT NO. 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

## 1.0 Introduction

By application dated March 17, 1978, as supplemented January 9, 1978, March 3 and 22, 1978, August 30, 1978, January 18, 1979, March 16, 1979, June 29, 1979, September 5, 1979, October 1 and 10, 1979, and December 5, 1979, the Florida Power Corporation (FPC) proposed to install high density fuel storage racks at the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3).

The proposed modification would increase the storage capacity in the spent fuel pools (SFPs) for up to 1153 fuel assemblies and six failed fuel containers. The proposed modification consists of replacing existing fuel assembly racks with high density, free standing storage racks without changing the basic structural geometry of the SFPs. (Two individual spent fuel storage pools are located within the fuel handling area of the Auxiliary Building. Both pools are rectangular: Pool A, 32' 2" by 24' 0" and Pool B, 32' 7" by 24' 0", with a depth of 43' 8".)

## 2.0 Discussion

Each storage rack consists of an assembly of fuel storage cells spaced 10.5 inches on center and welded to a base grid structure. Each storage cell is a double wall Type 304 stainless steel box with an inside square dimension of 8.9375 inches which allows sufficient clearance (0.2005 inch each side of the fuel assembly) to avoid interferences during fuel storage and removal operations. The double wall construction provides four compartments in which poison elements ($B_4C$ poison sheets) can be placed. The top opening of the storage cell is flared to facilitate insertion of the fuel assembly; the bottom member of the storage cell provides the level support surface required for the fuel assembly and contains the cooling flow orifice.

The bottom member of each storage cell sits on and is welded to the rack based unit which is basically a grid structure constructed from Type 304 stainless steel wide flange and box beam members. Continuous spacer bars are provided at the middle and top of the storage cells to ensure that the required pitch (10.5 inches) is maintained between storage cells in both directions (north/south and east/west). The spacer bars which are intermittently welded to the storage cells also maintain the vertical alignment of the cells. Support feet attached to the bottom of the rack base raise the rack above the pool floor to the height required to provide an adequately sized cooling water supply plenum (for natural circulation). Each support foot contains a remotely adjustable jackscrew to permit the rack to be leveled following installation.

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Because of the pool configurations (Pool A and Pool B) and requirements for failed fuel storage, it is necessary to supply different rack sizes. In addition to the basic 6 x 6 storage rack, racks are provided with 6 x 5, 5 x 5, 4 x 5, and 4 x 6 arrays of storage locations.

The storage racks which are free to slide are positioned on the pool floor so that adequate clearances are provided between racks and between the racks and pool structure to avoid impacting during seismic events. The horizontal seismic loads transmitted from the rack structure to the pool floor are only those associated with friction between the rack structure and the pool liner. The vertical dead-weight and seismic loads are transmitted directly to the pool floor by the support feet.

## 3.0 Evaluation

### 3.1 Structural and Mechanical

The supporting arrangements for the spent fuel modules, including their restraint, design, fabrication, and installation procedures; the structural design and analysis procedures for all loadings, including seismic and impact loadings; the load combinations; the structural acceptance criteria; the quality assurance requirements for design, fabrication, and installation; and applicable industry codes were all reviewed in accordance with the applicable portions of the NRC Position for Review and Acceptance of SFP Storage and Handling Applications of April 1978, as revised January 1979.

Seismic analysis was performed using pool floor response time histories which conform to those approved in the original plant design. The pool floor response time histories were determined in the seismic analysis of the Auxiliary Building using a base acceleration time-history compatible with smoothed response spectra which conform to the positions in Regulatory Guide 1.60, "Design Response Spectra for Seismic Design of Nuclear Power Plants," and structural damping values which conform to the positions in Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants." The pool floor horizontal time histories were then used as input to perform non-linear time-history analyses of the lateral motion of the fuel racks. The use of non-linear time-history analyses in the horizontal direction was necessitated by the non-linear characteristics of the fuel racks in the lateral direction. The combination of modes and spatial earthquake components in the seismic response analysis is in accordance with Regulatory Guide 1.92, "Combining Modal Responses and Spatial Components in Seismic Response Analysis".

In the Spent Fuel Rack Structural Analysis, the effects of a gap between a storage cavity and a fuel assembly, and the effects of submergence in water of the fuel racks were accounted for. The rack has been mathematically modeled as a three-dimensional finite-element structure consisting of discrete elastic beam and plate elements. Two representative fuel assembly load conditions (partially and fully loaded), were used in the analyses. The static analysis of the finite-element model has been performed using the direct stiffness methods of structural analysis to determine the internal forces and stresses in each element. For dynamic analyses, the natural frequencies and the mode shapes of the finite-element model were determined first, then the analyses were performed to determine the dynamic responses due to the seismic effects including the fuel impacting and hydrodynamic action. The total system response is obtained by combining the individual model response values in accordance with Regulatory Guide 1.92.

Non-linear time-history analyses were also performed to determine any potential impacting between adjacent fuel racks and between the fuel racks and the spent fuel pool structure. The storage rack and the stored fuel assemblies are represented by a two-dimensional lumped mass finite element model consisting of two finite-element cantilever beams, representing the storage cells and the stored fuel assemblies, attached to a floor mass by means of a non-linear sliding element. The range of friction coefficient used in the analyses, between the rack and two pool floor, was selected based on published test data. These analyses resulted in conservative values for the rack sliding and the shear forces transmitted through the rack and pool interfaces.

Rack material properties used in the analysis of the spent fuel racks are in accordance with the requirements of Subsection NF and Appendix I of Section III of the ASME Boiler and Pressure Vessel Code.

Results of the seismic analysis show that the racks are capable of withstanding the loads associated with all the design loading conditions without exceeding allowable stresses.

Analyses were performed to assess the effects of the fuel drop accidents. The postulated drop accidents include a straight drop on the top of a rack, a straight drop through an individual cell all the way to the bottom of the rack, and an inclined drop on the top of a rack. The drop height used in the analyses is 34 inches which is the maximum height that the crane can lift the fuel.

The SFPs are constructed of concrete walls and floors lined with stainless steel plates. Fuel pool structures have been analyzed for the additional loads resulting from the proposed increase in pool storage capacity and the most severe load combination conditions with the results indicating that the maximum loads are within the allowable stresses and the fuel pool floors are adequate to withstand the effects of the new racks and additional fuel.

## 3.2 Materials

The Type 304 stainless steel (ASTM Specification A-240) used in the new spent fuel storage racks is compatible with the storage pool environment, which is demineralized borated water controlled to a maximum 150°F temperature. To prevent any adverse effect from gas generated by $B_4C$ material exposure to the fuel pool environment, the poison material compartments are open at the top. Based on our review of previous operating experience with similar materials approved and in use, we have concluded that there is reasonable assurance that no significant corrosion of the racks, the fuel cladding, or the pool liner will occur over the lifetime of the plant.

## 3.3 Analysis, Design, Fabrication and Installation

The analysis, design, fabrication, and installation of the proposed new spent fuel rack storage system are in conformance with accepted codes and criteria. The analysis of the structural loads imposed by dynamic, static, seismic and thermal loadings; and the acceptance criteria for the appropriate loading conditions are in accordance with the appropriate portions of the NRC Position for Review and Acceptance of Spent Fuel Pool Storage and Handling Applications, April 1978, including errata, January 1979.

The mechanical properties for the materials used in the rack design are consistent with the normal and accident pool conditions. The quality assurance procedures for the materials, fabrication, installation, and examination of the new racks are in accordance with the accepted requirements of ASME Code, Section III, Subsection NF, Articles NF-2000, NF-4000, and NF-5000.

In addition, the design, procurement, and fabrication of the spent fuel racks comply with the pertinent requirements of Appendix B to 10 CFR 50, and delineated in Regulatory Guide 1.29, "Seismic Design Classification".

The effects of the additional loads on the existing pool structures due to the high capacity storage racks have been examined. The pool structural integrity is assured by conformance with the original Final Safety Analysis Report (FSAR) acceptance criteria.

There is no evidence at this time to indicate that corrosion of the fuel assemblies, the stainless steel rack structures, or the fuel pool liner will occur over the lifetime of the plant, at the temperatures and quality of the demineralized borated water to be maintained in the pools.

Installation procedures for the new racks have also been reviewed. Missile shields that are normally in place over the SFPs will remain in place over Pool B while old racks are being removed and new racks installed in Pool A. A similar procedure will be used for installation of new racks in Pool B. The licensee's analysis has shown that the missile shields will withstand the force of dropping a fuel rack from the maximum height that would be used in rack transfer operations. Based on handling procedures described to prevent damage to the stored fuel and to prevent interaction between old and new racks, the installation procedures have been found to be acceptable.

We found that the subject modification proposed by FPC is acceptable and satisfies the applicable requirements of the General Design Criteria 2, 4, 61, and 62 of 10 CFR, Part 50, Appendix A.

### 3.4 Criticality Consideration

The proposed spent fuel racks are to be made up of individual containers which are approximately 9 inches square by 14 feet long. These containers are to be fabricated from Type 304 stainless steel by using 1-1/4" x 1/8" angle stock for the corners which are welded to sides which consist of double sheets of .060" thick stock. Sheets of the Carborundum Company's Boron Carbide Composite Material, which are approximately 6.7 inches wide by 0.075 inches thick will be placed between these double sheets of stainless steel prior to welding. Since there will be a sheet of boron material in each of the double container walls, and since there will be one container for every fuel assembly, there will be two sheets of this boron material between every two fuel assemblies. Spacer grids and clips will be used to separate these containers in the modules to obtain a design lattice pitch of 10.5 inches. This will result in their being about one inch of water between the containers. This 10.5 inch pitch and the overall dimension of the fuel assembly, which is 8.52 inches, gives a fuel region volume fraction of 0.658 for the storage lattice.

FPC states that the highest anticipated Uranium-235 enrichment is 3.3 weight percent. This enrichment along with the technical specification limit on the loading of uranium dioxide in a fuel assembly, which is 536.94 kilograms, results in a maximum loading of 42.7 grams of Uranium-235 per axial centimeter of fuel assembly.

The FPC fuel pool criticality calculations are based on unirradiated fuel assemblies with no burnable poisons which have a fuel enrichment of 3.3 weight percent Uranium-235 and pure, i.e., unborated, water in the pool. 3.3 weight percent Uranium-235 corresponds to 42.7 grams of Uranium-235 per axial centimeter of fuel assembly with the present fuel. FPC also stated in its March 22, 1978 submittal that the areal density of the boron in each of the plates would be a minimum of 0.012 grams of boron-10 per square centimeter of plate and that this minimum amount of boron is used to calculate the neutron multiplication factors.

Nuclear Energy Services, Incorporated (NES) performed the criticality analyses for FPC. NES made parametric calculations by using the HAMMER computer program to obtain four-group cross sections for EXTERMINATOR diffusion theory calculations. The blackness theory program, BRM, was used to calculate the thermal and epithermal group cross sections for the boron region. This calculational method was used to determine the nominal $k_\infty$ and then the effects of design and fabrication tolerances, changes in temperature, voids in the pool water, and abnormal dislocations of fuel assemblies in the racks. NES also did verification calculations with the KENO Monte Carlo program with sixteen group Hansen-Roach cross sections. In its March 22, 1978 submittal, FPC stated that the overall result of all of these calculations is that, with an assumed calculational uncertainty of +0.01, the maximum, "worst case", abnormal neutron multiplication factor is 0.9356.

In its March 16, 1979 response to our request for additional information, FPC stated that it will perform a surveillance test on coupons of the $B_4C$/Polymer Composite plates to verify the continued presence of the boron in the plates in the pools over the complete life of the storage racks. In addition, FPC will perform an on-site neutron attenuation test to verify that there are no missing boron plates in the racks.

The results of the neutron multiplication factor calculations submitted by FPC are generally lower than the results from other methods for similar fuel pool storage lattices. By comparing FPC's results with those from other methods we have determined that an additional uncertainty of +0.01 needs to be added to FPC's maximum, "worst case", abnormal neutron multiplication factor of 0.9356; so that for practical purposes the maximum neutron multiplication factor in these racks for the specified fuel loading and boron plate loading is 0.95. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

Since the neutron multiplication factor could increase with the fuel loading, we have determined that the use of these storage racks should be prohibited for fuel assemblies that contain more than 42.7 grams of Uranium-235 per axial centimeter of fuel assembly. An appropriate Technical Specification has been established.

We find that all factors that could affect the neutron multiplication factor in the pools have been conservatively accounted for and that the maximum neutron multiplication factor in the pools with the proposed racks will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a SFP. This 0.95 acceptance criterion is based on the uncertainties associated with the calculational methods and provides sufficient margins to preclude criticality in the fuel. Accordingly, there is a Technical Specification which limits the effective neutron multiplication factor in each SFP to 0.95.

We find that when any number of the fuel assemblies, which FPC described in these submittals, having no more than 42.7 grams of Uranium-235 per axial centimeter of fuel assembly or equivalent are loaded into the proposed racks, the keff in the fuel pools will be less than the 0.95 limit. We also find that in order to preclude the possibility of the keff in the fuel pools from exceeding this 0.95 limit without being detected, the use of these high density storage racks will be prohibited for fuel assemblies that contain more than 42.7 grams of Uranium-235, or equivalent, per axial centimeter of fuel assembly. On the basis of the information submitted, and the keff and fuel loading limits stated above, we conclude that the design is acceptable from criticality consideration.

### 3.5 Spent Fuel Cooling

The licensed thermal power for CR-3 is 2452 MWt. FPC currently plans to refuel this reactor annually at which times about 59 of the 177 fuel assemblies in the core will be replaced. To calculate the maximum heat loads in the SFPs, FPC assumed a 150-hour time interval between reactor shutdown and the time when either the 59 fuel assemblies in the normal refueling or the 177 fuel assemblies in a full core offload are placed in the spent fuel pools. For this cooling time, FPC used the method given in the NRC Standard Review Plan 9.2.5 to calculate maximum heat loads of $16.7 \times 10^6$ BTU/hr for sixteen successive refuelings and $33.4 \times 10^6$ BTU/hr for the full core offload which fills the pool after sixteen refuelings have been performed.

The spent fuel cooling system consists of two pumps and two heat exchangers. Each pump is designed to pump 1500 gpm ($7.5 \times 10^5$ pounds per hour), and each heat exchanger is designed to transfer $8.75 \times 10^6$ BTU/hr from 129°F fuel pool water to 95°F closed cycle cooling water, which is flowing through the shell side of the heat exchanger at a rate of $7.5 \times 10^5$ pounds per hour.

FPC states that this system, with two pumps running, will be able to keep the spent fuel pool outlet temperature below 128°F through the sixteenth annual refueling. For cooling an offloaded full core, FPC's March 16, 1977 response to our request for additional information stated that the Decay Heat Removal System could be aligned to cool the SFPs by closing six valves in the spent fuel cooling system and opening seven valves in the Decay Heat Removal System. This system has two loops each of which is designed to remove $30 \times 10^6$ BTU/hr at a 140°F outlet temperature.

In regard to emergency makeup water for the SFPs, Section 9.3.2.8 of the FSAR states that the eight inch diameter pipe to the Decay Heat Removal System is designed to Seismic Class I criteria and that it connects the SFPs to the 420,000 gallon borated water storage tank.

By using the method given on pages 9.2.5-8 through 14 of the November 24, 1975 version of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than $10^7$ seconds, for a decay time of 150 hours, we find the FPC's maximum heat loads in the SFPs are conservatively high. We also find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pools from 256 to 1159 is $3.5 \times 10^6$ BTU/hr. This is the difference in peak heat loads for the present and the modified pools.

We find that with two pumps operating the SFP cooling system can maintain the fuel pool outlet water temperature below 128°F for the normal refueling offload that fills the pools. We find that the capacity of the Decay Heat Removal System is adequate for maintaining the spent fuel water temperature below 140°F for the full core offload that fills the pools.

Since both the SFP Cooling System and the Decay Heat Removal System are seismic Class I systems, it is highly unlikely that a single failure could result in a complete loss of SFP cooling.

However, if this did occur just after a full core offload, the maximum heatup rate of SFP water would be about 9°F/hr. Thus assuming that SFP water temperature was at its maximum of 140°F at the time of the loss of cooling, it would be more than eight hours before the pool would start to boil. We calculate that after boiling starts the required water makeup rate will be less than 70 gpm. We find that eight hours will be sufficient time to establish a 70 gpm makeup rate.

We find that the present cooling capacity for the CR-3 SFPs will be sufficient to handle the incremental heat load that will be added by the proposed modification. We also find that this incremental heat load will not alter the safety considerations of SFP cooling from that which we previously reviewed and found to be acceptable.

## 3.6 Installation of Racks and Fuel Handling

Because of the two refuelings at CR-3, there are 120 spent fuel assemblies in the pools. These fuel assemblies will be located in Pool B during the modification of Pool A and the missile shield over Pool B will prevent damage to any of the spent fuel assemblies in the unlikely event that a load is dropped during the change of racks in Pool A. A similar routine will be used for Pool B modifications in which all the fuel will be in Pool A.

We find that, because CR-3 has two separated SFPs and has a missile shield over the pools, FPC can adequately protect the spent fuel assemblies stored in the pools during the change of racks. After the racks are installed in the pools, the fuel handling procedures in and around the pools will be the same as those procedures that were in effect prior to the proposed modifications.

## 3.7 Fuel Handling

The CR-3 Technical Specifications prohibit loads greater than 2750 pounds (the nominal weight of a fuel assembly and handling tool) to be transported over spent fuel in the SFPs except for removal of old racks and installation of new racks in which case the missile shield must be over the fuel in the alternate pool. We have, therefore, concluded that the likelihood of any heavy load handling accident is sufficiently small that the proposed modification is acceptable.

The potential consequences of fuel handling accidents in the SFP area presented in the CR-3 Safety Evaluation Report (SER) dated July 1974 are not changed by the proposed modifications to the SFPs.

## 3.8 Occupational Radiation Exposure

We have reviewed FPC's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational radiation exposure for this operation is estimated by FPC to be about 8.5 man-rem. We consider this to be a reasonable estimate. This estimate represents a small fraction of the total man-rem burden from occupational exposure at the plant.

The estimated man-rem exposure to re-rack the pools is based on FPC's detailed breakdown of occupational exposure for each phase of the pools' modification. FPC considered the number of individuals performing a specific job, their occupancy time while performing this job and the average dose rate in the area where the job will be performed. The modification will be done in a dry pool (i.e., each pool will be drained) after decontamination of the pool has been performed by hydro-lasers followed by vacuuming and filtration of the final 6 inches of water on the pool floor. By using these techniques, we conclude that each pool modification will be done in a manner that will ensure as low as is reasonably achievable (ALARA) exposures to the occupational workers.

FPC has presented alternative plans for the disposal of the old racks which considered removing, decontaminating and crating intact racks vs. removing, decontaminating, cutting and packaging the small sections. FPC is considering three methods of disposal of the old racks: (1) shipping the racks whole to Barnwell for burial; (2) cutting them into small sections to reduce the volume and then shipping them to Barnwell for burial; or (3) crating the racks whole and shipping the intact racks to a vendor for further decontamination and scrapping. Taking into account alternative disposal costs and exposures, FPC will make the final decision as to the choice of disassembly and disposal of the old racks so that exposures will be kept to levels that are ALARA.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by FPC for dose rates in the SFP area from radionuclide concentrations in the pool water and the spent fuel assemblies. The spent fuel assemblies themselves will contribute a negligible fraction of the dose rates in the pool area because of the depth of water shielding the fuel. Consequently, the occupational radiation exposure resulting from the additional spent fuel in the pools represents a negligible burden. Based on present and projected operations in the SFP area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure will not affect FPC's ability to maintain individual occupational doses to ALARA and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in the SFPs will not result in any significant increase in doses received by occupational workers.

### 3.8 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. The waste treatment systems were evaluated in the SER dated July 1974. There will be no change in the waste treatment systems or in the conclusions of the evaluation of these systems as described in Section 11 of the SER because of the proposed modification.

## 4.0 CONCLUSION

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

Dated: November 17, 1980

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

# ENVIRONMENTAL IMPACT APPRAISAL BY THE OFFICE OF NUCLEAR REACTOR REGULATION

## SUPPORTING AMENDMENT NO.36 TO FACILITY OPERATING LICENSE NO. DPR-72

## FLORIDA POWER CORPORATION, ET AL.

## CRYSTAL RIVER UNIT NO. 3

## NUCLEAR GENERATING PLANT

## DOCKET NO. 50-302

### 1.0 Introduction

By application dated March 17, 1978, as supplemented January 9 1978, March 3 and 22, 1978, August 30, 1978, January 18, 1979, March 16, 1979, June 29, 1979, September 5, 1979, October 1 and 10, 1979, and December 5, 1979, the Florida Power Corporation (FPC) proposed to increase the total spent fuel storage capacity at Crystal River Unit No. 3 Nuclear Generating Plant, (CR-3).

The environmental impact of the existing CR-3 fuel storage pool was considered in the CR-3 Final Environmental Statement (FES) issued May 1973. The purpose of this appraisal is to evaluate any additional environmental effects of this proposed increase in storage capacity. The CR-3 spent fuel storage system is described in our concurrently issued Safety Evaluation.

### 2.0 Generic Environmental Impact Statement

A Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575, Volumes 1-3) was issued by NRC in August 1979. The NRC staff evaluated and analyzed alternatives for handling and storage of spent light water power reactor fuel with emphasis on long-range policy. Consistent with the long-range policy, the storage of spent fuel addressed in the FGEIS is considered to be interim storage to be used until the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion by licensees of onsite fuel storage capacity by modification of existing spent fuel pools (SFPs). By the date of issuance of the FGEIS (August 1979), 40 applications for SFP capacity expansions had been approved with the finding in each case that the environmental impact of the proposed increased storage was negligible. However, since there are variations in storage pool designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommended that licensing reviews be done on a case-by-case basis to resolve plant-specific concerns. This appraisal accomplishes that recommendation.

In addition to the alternative of increasing the storage capacity of the existing SFPs, other spent fuel storage alternatives are discussed in detail in the FGEIS. The finding of the FGEIS is that the environmental impact-costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of the various alternatives reflect the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, where spent fuel generation is terminated, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

## 3.0 Need for Increased Storage Capacity

CR-3 is an 825 MWe nuclear power plant. Two fuel storage pools are provided. Currently there are 240 storage spaces in the SFPs for CR-3. CR-3 has 177 assemblies in its core.

The proposed increase would be accomplished by replacing the existing spent fuel storage racks with new, more compact, neutron absorbing racks. The proposed rack design is discussed in the concurrently issued SER. This modification would extend spent fuel storage capacility through 2002 compared to 1983 with the existing capacity. A more immediate concern is that of maintaining sufficient room in the SFP to off-load a full core (177 fuel assemblies) should this be necessary for inspection or repair of reactor internal equipment or piping. While this capability is not necessary to protect the health and safety of the public, it is desirable to reduce occupational exposures. With the present SFP capacity for CR-3, FPC does not now have full core discharge capability.

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shutdown in 1972 for alternations and expansion; on September 22, 1976, NFS informed the Commission that they were withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate.

The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois is in a decommissioned conditon. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois, and the storage pool at West Valley, New York (on land owned by the State of New York and leased to NFS through 1980) are licensed to store spent fuel. The storage pool at West Valley is not full but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractural arrangements with NFS. GE is also not accepting any additional spent fuel for storage at the MO. Construction of the AGNS receiving and storage station has been completed. AGNS has applied for, but has not been granted, a license to receive and store irradiated fuel assemblies in the storage pool at Barnwell prior to a decision on the licensing action relating to the separation facility.

## 4.0 Radioactive Wastes

CR-3 contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the CR-3 FES dated May 1973. There will be no change in the waste treatment systems described in Section 3.4.2 of the FES because of the proposed modification.

## 5.0 SFP Cleanup System

The SFP cleanup system consists of two cartridge filters and an ion exchanger and the required piping, valves and instrumentation. This system is in parallel to the two SFP cooling loops in the SFP cooling system. Each of the two SFP pumps draws water from the SFPs, circulates it through a heat exchanger and the SFP cleanup system and returns it to the SFPs. The cleanup system may be bypassed manually if required.

Because we expect only a small increase in radioactivity released to the pool water as a result of the proposed modification as discussed in Section 6.2, we conclude that the SFP purification system will keep concentrations of radioactivity in the pool water to levels which have existed prior to the modification.

## 6.0 Environmental Impacts of the Proposed Action

### 6.1 Nonradiological

The environmental impact of CR-3 as designed, was considered in the FES. Increasing the number of assemblies stored in the existing fuel pools will not cause any new environmental impacts. The amount of waste heat emitted by CR-3 will increase slightly (less than one percent), resulting in no measurable increase in impact upon the environment.

### 6.2 Radiological

#### Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity were evaluated and determined to be environmentally insignificant as addressed below.

The additional spent fuel which would be stored due to the expansion is the oldest fuel which has not been shipped from the plant. This fuel should have decayed about three years. During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominately nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution to radiation levels to which workers in and near the SFPs would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFPs during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFPs. During and after refueling, the SFP cleanup system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800 F. A few weeks after refueling, the spent fuel cools in the SFPs so that fuel clad temperature is relatively cool, approximately 180 F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap.

In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on the operational reports submitted by the licensees or discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the MO (formerly Midwest Recovery Plant), or at NFS storage pool at West Valley, New York. Spent fuel has been stored in these two pools which, while it was in a reactor, was determined to have significant leakage and was therefore removed from the core. After storage in the onsite SFP, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

### 6.2.1 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defected fuel. However, we have conservatively estimated that less than an additional 40 curies per year of Krypton-85 may be released from the SFPs for CR-3 when the modified pools are completely filled. This increase would result in an additional total body dose of less than 0.0001 mrem/year to an individual at the site boundary. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less than 0.0001 man-rem/year. This is small compared to the fluctuations in the annual dose this population would receive from natural background radiation. This exposure represents an increase of less than 0.5% of the exposure from the plant evaluated in the FES in Tables 5.10 and 5.11. Thus, we conclude that the proposed modification will not have any significant impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refuelings.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature during normal refuelings above the 120 F used in the design analysis in the FSAR. Therefore, it is not expected that there will be any significant change in the annual release of tritium or iodine as a result of the proposed modification from that previously evaluated in the FES.

Most airborne releases from the plant result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than the SFPs. Therefore, even if there were a slightly higher evaporation rate from the SFPs, the increase in tritium and iodine released from the plant as a result of the increase in stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FES. If levels of radioiodine become too high, the air can be diverted to charcoal filters for the removal of radioiodine before its release to the environment. The plant radiological effluent Technical Specifications, which are not being changed by this action, restrict the total gaseous releases of radioactivity from the plant including the SFPs.

### 6.2.2 Solid Radioactive Wastes

The concentration of radionuclides in the pools is controlled by the filter and ion exchanger and by decay of short-lived isotopes. The activity is high during refueling operations while reactor coolant water is introduced into the pools and decreases as the pool water is processed through the filter and ion exchanger. The increase of radioactivity, if any, should be minor because the additional spent fuel to be stored is relatively cool, thermally, and radionuclides in the fuel will have decayed significantly.

While we believe that there should not be an increase in solid radwaste due to the modification, as a conservative estimate, we have assumed that the amount of solid radwaste may be increased by 42 cubic feet of resin a year from the demineralizer (two additional resin bed/year). Because CR-3 has been in commercial operation only since 1977, we do not believe the record of solid waste shipped from the plant may be representative. The annual average amount of solid waste shipped from a single pressurized water reactor between 1974 and 1976 was about 10,000 cubic feet per year. If the storage of additional spent fuel does increase the amount of solid waste from the SFP purification system by about 42 cubic feet per year, the increase in total waste volume shipped would be less than 0.5% and would not have any significant environmental impact.

The present spent fuel racks to be removed from the SFPs are contaminated and will either be disposed of as low level waste or sent to a vendor to be salvaged. FPC has stated that less than 6,300 cubic feet of solid radwaste will be removed from the SFPs because of the proposed modification. This is if the racks are not cut into small pieces. Therefore, the total waste shipped from the plant would be increased by less by less 1.6% per year when averaged over the lifetime of the plant. This additional low level waste will not have any significant environmental impact. The plant radiological effluent Technical Specifications, which are not being changed by this action, restrict the total liquid releases of radioactivity from the plant.

### 6.2.3 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. The amount of radioactivity on the SFP filter and demineralizer might slightly increase due to the additional spent fuel in the pools, but this increase of radioactivity should not be released in liquid effluents from the plant. The plant radiological effluent Technical Specifications, which are not being changed by this action, restrict the total liquid releases of radioactivity from the plant.

The cartridge filter removes insoluble radioactive matter from the SFP water. This is periodically removed to the waste disposal area in a shielded cask and placed in a shipping container. The insoluble matter will be retained on the filter or remain in the SFP water.

The demineralizer resins are periodically flushed with water to the spent resin storage tank. The water used to transfer the spent resin is decanted from the tank and returned to the liquid radwaste system for processing. The soluble radioactivity will be retained on the resins. If any activity should be transferred from the spent resin to this flush water, it would be removed by the liquid radwaste system.

Leakage from the SFPs is collected in the Auxiliary Building Sump. This water is transferred to the liquid radwaste system and is processed by the system before any water is discharged from the plant.

Visual observations can be made to determine if there are leaks in the SFP liner. Monitoring equipment will alarm in the control room if the pool water level falls below a predetermined level. To date, no water leakage from the SFPs has been observed.

6.2.4 Occupational Radiation Exposures

We have reviewed FPC's plans for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. The occupational exposure for the entire operation is estimated by FPC to be about 8.5 man-rem. We consider this to be a reasonable estimate because it is based on realistic dose rates and occupancy factors for individuals performing a specific job during pool modification. This operation is expected to be a small fraction of the total annual man-rem burden from occupational exposure.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by FPC for dose rates in the SFP area from radionuclide concentrations in the SFP water and from the spent fuel assemblies. The spent fuel assemblies themselves will contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. Consequently, the occupational radiation exposure resulting from the additional spent fuel in the pools represents a negligible burden. Based on present and projected operations in the SFP area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation exposure burden at this facility. Thus, we conclude that storing additional fuel in the SFPs will not result in any significant increase in doses received by occupational workers.

6.2.5 Impacts of Other Pool Modifications

As discussed above, the additional environmental impacts in the vicinity of CR-3 resulting from the proposed modification are a very small fraction (less than 1%) of the impacts evaluated in the CR-3 FES. These additional impacts are too small to be considered anything but local in character.

Based on the above, we conclude that an SFP modification at any other facility should not significantly contribute to the environmental impact of the CR-3 and that the CR-3 SFP modification should not contribute significantly to the environmental impact of any other facility.

7.0 Environmental Impact of Postulated Accidents

Although the new high density racks will accommodate a larger inventory of spent fuel, we have determined that the installation and use of the racks will not change the radiological consequences of a postulated fuel handling accident in the SFP area from those values reported in the CR-3 FES dated dated May 1973.

Additionally, the NRC staff has under way a generic review of load handling operations in the vicinity of SFPs to determine the likelihood of a heavy load impacting fuel in the pool and, the radiological consequences of such an event. FPC is required to prohibit loads greater than 2,750 pounds (the nominal weight of a fuel assembly and handling tool) to be transported over spent fuel in the SFPs, except during the removal of old racks and placement of new racks in the pools. During this activity the fuel will be in the alternate pool which will be covered with missile shields. We have, therefore, concluded that the likelihood of any load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the SFPs are necessary while our review is under way.

8.0 Radiological Impact on Environment

As discussed in Section 6, expansion of the storage capacity of the SFPs will not create any significant additional radiological effects. The additional total body dose that might be received by an individual or the estimated population within a 50-mile radius is less than 0.0001 mrem/yr and 0.0001 man-rem/yr, respectively. These exposures are small compared to the fluctuations in the annual dose this population receives from background radiation. The population exposure represents an increase of less than 0.5% of the exposures from the plant evaluated in the FES. The occupational radiation exposure of workers during removal of the present storage racks and installation of the new racks is estimated by FPC to be about 8.5 man-rem. This is a small fraction of the total man-rem burden from occupational exposure at the plant. Operation of the plant with additional spent fuel in the SFPs is not expected to increase the occupational radiation exposure by more than one percent of the present total annual occupational exposure at this facility.

## 9.0 Basis and Conclusion for Not Preparing an Environmental Impact Statement

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action other than has already been predicted and described in the Commission's FES for CR-3. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

Dated: November 17, 1980

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-302

FLORIDA POWER CORPORATION, ET AL.

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 36 to Facility Operating License No. DPR-72, issued to the Florida Power Corporation, City of Alachua, City of Bushnell, City of Gainesville, City of Kissimmee, City of Leesburg, City of New Smyrna Beach and Utilities Commission, City of New Smyrna Beach, City of Ocala, Orlando Utilities Commission and City of Orlando, Sebring Utilities Commission, Seminole Electric Cooperative, Inc., and the City of Tallahassee (the licensees) which revised the Technical Specifications for operation for the Crystal River Unit No. 3 Nuclear Generating Plant (the facility) located in Citrus County, Florida. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specifications to authorize an increase in the capacity of the spent fuel storage pools at the facility.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on June 15,

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1978 (43 FR 25885). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an Environmental Impact Appraisal for this action 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.

For further details with respect to this action, see (1) the application for amendment dated March 17, 1978, as supplemented, (2) Amendment No. 36 to License No. DPR-72, (3) the Commission's related Safety Evaluation and (4) the Commission's 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), (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 17th day of November 1980.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

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