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

April 16, 1991

Docket No. 50-302

Mr. Percy M. Beard, Jr.
Senior Vice President,
Nuclear Operations
Florida Power Corporation
ATTN: Manager, Nuclear Operations
Licensing
P. O. Box 219-NA-2I
Crystal River, Florida 32629

Dear Mr. Beard:

SUBJECT: CRYSTAL RIVER UNIT 3 - ISSUANCE OF AMENDMENT RE: SPENT FUEL EXPANSION (TAC NO. 75305)

The Commission has issued the enclosed Amendment No. 134 to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). This amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 31, 1989, as supplemented by letters dated January 25, March 8, June 21, August 23, November 8, and November 28, 1990.

This amendment revises the TS to permit an increase in the capacity of spent fuel storage pool B and in the allowable fuel enrichment in fuel pool B.

Your application included a one-time relief allowing removal of the missile shields over spent fuel pool B while modifying fuel racks. This relief is no longer necessary, and, as agreed to in discussions with your staff, is not part of this amendment. The Notice of Withdrawal of this portion of the application for amendment is enclosed.

The application also included TS in both the current and new format of the Technical Specification Improvement Program. Only the current format TS changes have been reviewed and approved by the staff in this amendment.

A copy of the Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

NRC FILE CENTER COPY

This completes our efforts on TAC No. 75305.

Sincerely,

(Original signed by)

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No.134 to DPR-72
2. Safety Evaluation
3. Notice of Withdrawal

cc w/enclosures:
See next page

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AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-72-CRYSTAL RIVER UNIT 3

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S. Varga, 14/E/4
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H. Berkow
D. Miller
H. Silver
L. Kopp 8/E/23
S.B. Kim 7/H/15
N. Wagner 8/D/1
J. Wang 10/D/4
S. Parczewski 7/D/4
R. Jones 8/E/23
G. Bagchi 7/H/15
C. McCracken 8/D/1
L. Cunningham 10/D/4
OGC-WF
D. Hagan, 3302 MNBB
E. Jordan, 3302 MNBB
B. Grimes, 9/A/2
G. Hill (4), P-137
Wanda Jones, P-130A
J. Calvo, 11/F/23
J. Miller, 11/F/23
ACRS (10)
GPA/PA
OC/LFMB
M. Sinkule, R-II

cc: Plant Service list

Mr. Percy M. Beard, Jr.
Florida Power Corporation

Crystal River Unit No. 3 Nuclear
Generating Plant

cc:
Mr. A. H. Stephens
General Counsel
Florida Power Corporation
MAC - A5D
P. O. Box 14042
St. Petersburg, Florida 33733

Mr. P. F. McKee, Director
Nuclear Plant Operations
Florida Power Corporation
P. O. Box 219-NA-2C
Crystal River, Florida 32629

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
1700 Rockville Pike, Suite 525
Rockville, Maryland 20852

Senior Resident Inspector
Crystal River Unit 3
U.S. Nuclear Regulatory Commission
6745 N. Tallahassee Road
Crystal River, Florida 32629

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street N.W., Suite 2900
Atlanta, Georgia 30323

Mr. Jacob Daniel Nash
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

State Planning and Development
Clearinghouse
Office of Planning and Budget
Executive Office of the Governor
The Capitol Building
Tallahassee, Florida 32301

Chairman
Board of County Commissioners
Citrus County
110 North Apopka Avenue
Inverness, Florida 32650

Mr. Rolf C. Widell, Director
Nuclear Operations Site Support
Florida Power Corporation
P.O. Box 219-NA-2I
Crystal River, Florida 32629

Mr. Gary L. Boldt
Vice President, Nuclear Production
Florida Power Corporation
P. O. Box 219-SA-2C
Crystal River, Florida 32629

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. 134
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 October 31, 1989, as supplemented by letters dated January 25, March 8, June 21, August 23, November 8, and November 28, 1990, 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.

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:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 134, 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 its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 16, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 134

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

Remove	Insert
3/4 9-13	3/4 9-13
-----	3/4 9-13a
-----	3/4 9-15
5-5	5-5
5-6	5-6

3/4.9.13 FUEL ASSEMBLY STORAGE

LIMITING CONDITION FOR OPERATION

3.9.13 Fuel assemblies shall be stored in locations in accordance with conditions as specified below:

LOCATION	CONDITION
1. Dry fuel storage racks	New fuel with initial enrichment ≤ 4.5 weight percent U-235
2. Storage Pool A	Fuel with initial enrichment and burnup in accordance with Figure 3.9-1 but ≤ 4.5 weight percent U-235
3. Storage Pool B: Region 1	Fuel with initial enrichment ≤ 4.2 weight percent U-235
4. Storage Pool B: Region 2	Fuel with initial enrichment and burnup in accordance with Figure 3.9-2 but ≤ 4.2 weight percent U-235

APPLICABILITY: Whenever fuel assemblies are in the fuel storage locations.

ACTION:

a. With the requirements of the above specification not satisfied, suspend all other movement of fuel assemblies and crane operations with loads in the fuel storage areas and move the non-complying fuel assemblies to their proper designated locations. In addition, with the requirements of the above specification for storage pools A or B not satisfied, boron concentration of the spent fuel pools shall be verified to be greater than or equal to 1925 ppm at least once per 8 hours.

b. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.13.1 Verify the initial enrichment of new fuel assemblies ≤ 4.5 weight percent U-235 prior to storage in the dry fuel storage racks.

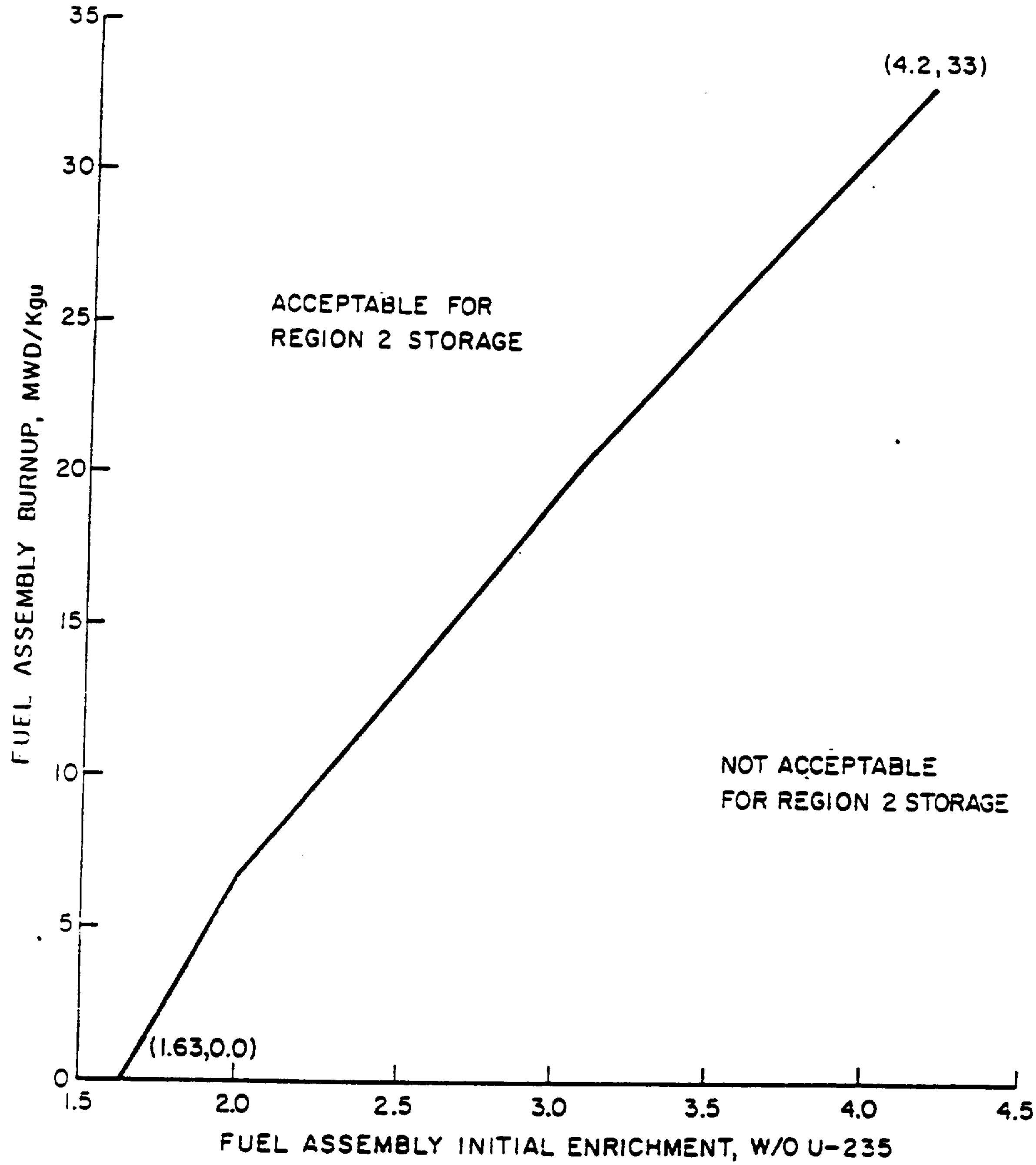
4.9.13.2 Perform an INDEPENDENT VERIFICATION of the initial enrichment and burnup of fuel assemblies in accordance with Figure 3.9-1, prior to storage in the pool A storage racks. A complete record of such analysis shall be kept for the time period that the fuel assembly remains in the pool A storage racks.

4.9.13.3 Verify the initial enrichment of fuel assemblies ≤ 4.2 weight percent U-235 prior to storage in Region 1 of pool B storage racks.

4.9.13.4 Perform an INDEPENDENT VERIFICATION of the initial enrichment and burnup of fuel assemblies in accordance with Figure 3.9-2, prior to storage in Region 2 of pool B storage racks. A complete record of such analysis shall be kept for the time period that the fuel assembly remains in the pool B storage racks.

FIGURE 3.9-2

MINIMUM REQUIRED FUEL ASSEMBLY BURNUP AS A FUNCTION OF INITIAL ENRICHMENT TO PERMIT STORAGE IN REGION 2 OF POOL B



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 dry fuel storage racks and spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} less than or equal to 0.95 when flooded with unborated water, which includes a conservative allowance for uncertainties. This is based on new fuel with a maximum initial enrichment of 4.5 weight percent U-235 in dry fuel storage racks, on fuel with combinations of initial enrichment and discharge burnup as shown in Figure 3.9-1 in storage pool A, and on fuel with a maximum initial enrichment of 4.2 weight percent in Region 1 of storage pool B, and on fuel with combinations of initial enrichment and discharge burnup as shown in Figure 3.9-2 in Region 2 of storage pool B;

b. A nominal 10.5 inch center-to-center distance between fuel assemblies placed in the high density storage racks in pool A; and

c. A nominal 21.125 inch center-to-center distance between fuel assemblies placed in the dry fuel storage racks; and

d. A nominal 10.6 inch center-to-center distance between fuel assemblies placed in Region 1 of spent fuel pool B; and a nominal 9.17 inch center-to-center distance between fuel assemblies placed in Region 2 of spent fuel pool B.

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.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1357 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. 134 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 letter dated October 31, 1989, as supplemented by letters dated January 25, March 8, June 21, August 23, November 8, and November 28, 1990, Florida Power Corporation (FPC or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. DPR-72 for the Crystal River Unit No. 3 Nuclear Generating Plant (CR-3). The proposed amendment would provide:

1. a one-time relief from TS 3.9.11 to allow removal of the missile shields for installation of high density spent fuel storage racks in pool B. This one-time relief would be in effect for the duration of the spent fuel pool B rerack modification. However, this proposed change is no longer necessary and is not addressed further herein.

2. an increase in the allowable nominal fuel enrichment from 4.0 to 4.2 weight percent of U-235 for spent fuel pool B.

3. an increase in the number of spent fuel storage locations to 1357 for both pools, resulting from an increase in spent fuel storage locations in pool B to 815, and a decrease in the number of failed fuel containers to 0 in pool B.

4. an expansion of TS 5.6.1 to indicate that the high-density spent fuel racks in pool B will utilize a two region layout. Region 1 will have a 10.60 inch center-to-center spacing and Region 2 will have a 9.17 inch center-to-center spacing.

The letters dated March 8, June 21, August 23, November 8 and November 28, 1990 provided supplemental information which did not change the initial proposed no significant hazard consideration determination.

2.0 DISCUSSION

There are two spent fuel pools at CR-3. Spent fuel pool A contains high-density storage racks, while spent fuel pool B utilizes standard racks. Pool B contains storage space for 120 spent fuel assemblies. With the present spent fuel storage capacity, CR-3 lost full core reserve storage after Refuel VII in 1990. Therefore, the licensee has proposed to replace the existing pool B storage racks with new high-density, free-standing spent fuel storage racks which will allow for more dense storage of spent fuel by utilizing a neutron absorbing material (Boraflex). Fuel storage will be divided into two regions within pool B. Region 1 (174 locations) will have a fuel assembly center-to-center spacing of 10.60 inches, while Region 2 (641 locations) will have a 9.17 inch center-to-center spacing. Region 1 is designed to accommodate either irradiated or non-irradiated, 4.2 weight percent (w/o) U-235 enriched fuel from core off-loads. Region 2 is designed to accommodate irradiated fuel which has achieved sufficient burnup. The racks are also designed to store fuel consolidated at a 2:1 consolidation ratio. The new racks will permit full core offloading through the year 2008. All failed fuel containers in pool B will be removed.

The spent fuel pool is designed as a Seismic Category I structure. Two individual spent fuel storage pools are located within the fuel handling area of the Auxiliary Building. Both pools are rectangular in plan and both pools have a depth of 43' - 8". The walls and bottom slab of the spent fuel pools have nominal thickness of 5' - 0". Wetted surfaces of the pools are lined with stainless steel to ensure watertight integrity. The pools are filled with air-saturated demineralized water with 1925 ppm boron.

The spent fuel storage pools are supported on reinforced concrete walls extending downward to the top of the structural foundation mat. The spent fuel pool is enclosed within the limits of the steel-framed and metal-sided Auxiliary Building structure.

The storage racks and individual storage cells will be constructed of Type 304LN stainless steel with leveling screws made from Type 17-4 PH stainless steel. The Boraflex neutron absorbing material to be used in the new racks is a silicone-based polymeric material containing fine boron carbide particles dispersed in a homogeneous matrix, with panels 7.5 inches wide and 0.085 inches thick held at the side of the cell by stainless steel wrappers. The wrappers are attached to the outside of the cells by spot welding the entire length of the wrapper. There is a gap of 0.120 inches between the cell wall and the wrapper, providing an unrestrained shrinkage for Boraflex panels during irradiation. Vent holes are provided in the wrapper for venting gases formed during irradiation.

The licensee proposed a long-term surveillance program to monitor performance of the Boraflex in the spent fuel pool environment. Visual examination capability will be provided in the cell wall in each rack and surveillance coupons will be used to determine effects of gamma radiation. The initial surveillance will be implemented after approximately 2 years of exposure and will consist of visual inspection as well as other tests required for verifying performance of the Boraflex. Future surveillance will be based on the results of these initial examinations. Should the Boraflex show signs of degradation which will impair its neutron absorption capability beyond the limits assumed in the criticality analysis, the licensee will be required to take corrective actions.

3.0 EVALUATION

3.1 Reactor Systems

The analysis of the reactivity effects of fuel storage in Region 1 was performed with the KENO-IV three-dimensional Monte Carlo computer code. For Region 2, burnup-dependent reactivity calculations were performed with the depletable, two-dimensional, transport theory code PHOENIX, whereas KENO-IV was used for the nominal case reactivity. These codes are widely used for the analysis of fuel rack reactivity and have been benchmarked against results from numerous critical experiments. The staff concludes that the analysis methods used are acceptable.

The analyses were performed with several assumptions which tend to maximize the rack reactivity. These include:

(1) unborated pool water at a temperature yielding the highest reactivity

(2) use of the B&W 15x15 fuel assembly at its most reactive point in life and at the highest enrichment authorized without any control rods or burnable poisons

(3) assumption of infinite extent in lateral direction

(4) absorption effect of the fuel assembly spacer grids is neglected

(5) minimum boron loading in Boraflex absorber material.

The staff concludes that appropriately conservative assumptions were made.

For the nominal storage cell design in Region 1, uncertainties due to manufacturing tolerances were treated by either using worst-case conditions or by performing sensitivity studies. These include variations in poison pocket thickness, stainless steel thickness, cell ID, center-to-center spacing, and cell bowing. These uncertainties were appropriately determined at least at the 95 percent probability, 95 percent confidence (95/95 probability/confidence) level. In addition, a method bias and uncertainty were determined from benchmark calculations. The final Region 1 design, when fully loaded with fuel enriched to 4.2 w/o U-235, resulted in a k_{eff} of 0.9499 when combined with all known uncertainties. This meets the staff's criterion of k_{eff} no greater than 0.95, including all uncertainties at the 95/95 probability/confidence level and is, therefore, acceptable.

For Region 2, the same uncertainties were considered. In addition, an uncertainty associated with the burnup-dependent reactivities computed with PHOENIX was accounted for in the development of the Region 2 burnup requirements. A series of reactivity calculations was made to generate a set of enrichment fuel assembly discharge burnup ordered pairs which all yield the equivalent k_{eff}. This method for obtaining the constant reactivity curve for required burnup as a function of enrichment is the standard one used for rack reactivity evaluations and is acceptable. TS Figure 3.9-2 shows the constant k_{eff} contour generated for the CR-3 Region 2 racks. From this figure, it can be seen that the reactivity of the racks containing fuel at 33,000 MWD/MTU burnup which had an initial enrichment of 4.2 w/o U-235 is equivalent to the rack reactivity with fresh fuel (zero burnup) having an initial enrichment of 1.63 w/o U-235. This configuration resulted in an acceptable k_{eff} of 0.9408, including all appropriate uncertainties.

The storage racks are also capable of storing an array of consolidated fuel rods in which the rods from two fuel assemblies are packed into one assembly (consolidation ratio of 2:1) in each storage location. Analyses have shown that if a consolidation ratio of 2:1 is maintained, the consolidated array is always less reactive than the unconsolidated array if the same type of fuel is stored in the same rack cell.

Most abnormal storage conditions will not result in an increase in the k_{eff} of the racks. However, it is possible to postulate events, such as the misloading of an assembly with a burnup and enrichment combination outside of the acceptable area in TS Figure 3.9-2, or dropping an assembly between the pool wall and the fuel racks, which could lead to an increase in reactivity. However, for such events, credit may be taken for the presence of approximately 2,000 ppm of boron in the pool water since the staff does not require the assumption of two unlikely, independent, concurrent events to ensure protection against a criticality accident (double contingency principle). The reduction in k_{eff} caused by the boron more than offsets the reactivity addition caused by credible accidents.

The following TS changes have been proposed as part of the requested spent fuel pool reracking. The staff finds these changes acceptable.

(1) TS 3.9.13 includes an increase in the allowable nominal fuel enrichment to 4.2 w/o for spent fuel pool B and references Region 1 and Region 2.

(2) Figure 3.9-2 has been modified to specify the minimum required burnup as a function of initial enrichment to permit storage in Region 2 of pool B.

(3) TS 5.6.1 specifies the center-to-center distance between fuel assemblies in Region 1 and Region 2 of fuel pool B.

(4) TS 5.6.3 specifies an increase in the number of spent fuel storage locations from 1153 to 1357 for both pools.

Based on the review described above, the staff finds that the criticality aspects of the proposed modifications to the CR-3 spent fuel pool storage racks are acceptable and meet the requirements of General Design Criterion 62 for the prevention of criticality in fuel storage and handling. The staff concludes that fuel from CR-3 may be safely stored in Region 1 of pool B provided that the U-235 enrichment does not exceed 4.2 w/o. Any of these fuel assemblies may also be stored in Region 2 of pool B provided they meet the burnup and enrichment limits specified in Figure 3.9-2 of the CR-3 TS.

3.2 Structural Engineering

This evaluation addresses the adequacy of the structural seismic aspects of the application in support of the proposed license amendment on the use of high-density spent fuel racks at CR-3. The primary areas of review associated with the proposed application are focused towards assuring the structural integrity of the fuel, fuel cells, rack modules, and the spent fuel pool floor and walls under the postulated loads (Appendix D of SRP 3.8.4, NUREG-0800) and fuel handling accidents.

3.2.1 Spent Fuel Storage Pool

The licensee compared concrete strength capacities with anticipated loads on the concrete structure from high-density rack dynamic loads (operating basis earthquake (OBE), safe shutdown earthquake (SSE), and fuel drop) and margins were shown to be available with respect to allowable stresses. The licensee stated that the pool structural integrity is assured by conformance with the FSAR acceptance criteria.

The staff, therefore, concludes that the CR-3 spent fuel pool would continue to support loads due to additional fuel and new fuel racks for normal as well as accident conditions.

3.2.2 Fuel Handling Accidents

Two cases were considered for the accidental drop of a fuel assembly; a B&W 15x15 standard fuel assembly of dry weight 2250 pounds and consolidated fuel assembly of dry weight 4100 pounds, both with handling tool and drop height of 24 inches. The kinetic energy of the fuel assembly is totally converted into strain energy of the assembly structure and the result indicated that fuel criticality does not occur. Also, the result has shown that, when considering the bottom nozzle of the fuel impacting on the floor of the pool, the stress imposed on the pool liner was determined to be less than the ASME Code allowable limit for faulted condition. Therefore, the licensee concluded that the liner will not be perforated for any of the drop accidents.

The licensee also investigated a case for a jammed fuel assembly during spent fuel handling machine uplift operation which generates an uplift load of 350 pounds. The stresses resulting from this load were calculated and found to be acceptable when compared to the acceptance limits. Therefore, the licensee's accident analysis is acceptable.

3.2.3 Fuel Rods

Prior to discharge of depleted fuels to the spent fuel pool, the fuel assembly, which consists of 15x15 array of fuel rods and spacer grids, and other components of the assembly, undergo up to 3 years of burn-up. As a part of the operating license application process, several safety analyses were performed and the staff reviewed and approved the integrity of the fuel assembly for normal operating, transient and accident conditions, such as LOCA and design basis earthquake. One of the accident conditions considered in the FSAR and the staff Safety Evaluation Report is the combined effect of the seismic and LOCA dynamic loads on fuel assemblies. The dynamic effect of a LOCA is combined with that of the earthquake and their resulting forces are applied to the fuel assembly via reactor core plate motion. In the reactor, the fuel and fuel assembly do not have protection from fuel cells as is the case when they are stored in the spent fuel pool. A fuel cell is an individual cell that surrounds a fuel assembly and is part of the rack. The elevation of the fuel in the reactor is equivalent to that of the spent fuel pool. Therefore, the fuel movement and, hence, the dynamic motion of the fuel in the spent fuel pool due to earthquake loads would be similar to that of the fuel in the reactor, thus providing assurance that fuel rods evaluated for in-reactor seismic-LOCA dynamic analysis would survive seismic motion in the spent fuel pool.

3.2.4. Rack Modules

The rack modules are composed of individual storage cells made of stainless steel. The inside width of the cell is 9 inches, allowing B&W 15x15 or consolidated fuel assemblies to slip in with enough gap left to allow differential thermal expansion. The rack array is typically 10X11, thus accommodating approximately 110 fuel assemblies per rack. The average rack has a width-to-depth ratio of 90 inches to 100 inches with a height of 167 inches. Each rack module is provided with leveling pads which contact the spent fuel pool floor. The spent fuel storage racks are Seismic Category I equipment and, therefore, they are required to remain functional during and after an SSE. They are neither anchored to the pool floor or to the pool wall, nor structurally interconnected.

The licensee compared the seismically-induced stresses against design limits to ensure the structural adequacy of the design. The horizontal displacement results were compared with the rack clearances to determine whether the racks collide with another rack or the pool wall. The vertical displacements of the support pads, due to rack rocking and lift-off, were used to show that the racks do not overturn. The results showed that the high density spent fuel racks are structurally adequate to resist the postulated stress combinations and the racks have margin against overturning and against rack collision.

Calculation of stresses, displacement and dynamic stability of the rack was performed by time history analyses on the nonlinear dynamic models using the dynamic capabilities of the nonlinear modal superposition solution technique of the Westinghouse Electric Computer Analysis (WECAN) Code. The nonlinear modal superposition method was developed to analyze nonlinear structural dynamics problems involving impact between components and Coulomb friction. The finite element method is used to express the equations of motions with the nonlinearities represented by a pseudo force vector.

Modal superposition was possible because nonlinear stiffness and damping terms are transposed to right hand side of the equal sign as a pseudo force term and, hence, the left hand side of the equal sign consists only of linear terms. Pseudo force terms are in turn expanded into a Taylor Series to further simplify the calculation algorithm. The staff had some concern about the way the Taylor Series expansion was applied in this case. In particular, the pseudo force terms are functions of several variables, such as displacement and velocity, whereas the series was expanded in terms of time only. Implication of time dependent expansion on the final result disregarding other dependent variables was not clear. The licensee did not demonstrate explicitly why such a limited expansion is adequate. The licensee also did not perform a rigorous error assessment in terms of numerical analysis, or the confirmation of its result with physical data as far as nonlinear dynamic response is concerned. For this reason, the staff performed the following independent assessment of the rack design as described below.

In a diagram provided by the licensee, it is indicated that the typical gap between rack modules and the spent fuel pool wall is 8 inches or less, except that one module in Region I has 16.6 inches on one side of the module. Between the rack modules, the space is less than 2 inches. With this geometrically constrained configuration it is doubtful the rack modules would develop enough kinetic energy during an SSE so as to damage the spent fuel pool steel liner or rack module themselves. Damage to fuel rods, whose function is to contain fission products, has been discussed already in the previous section. All the vendor's analyses which the staff has reviewed stated that impact between the racks and rack and wall is not expected. Analysis for CR-3 also predicted that there will be no impact during an SSE. Even if the steel liners were to be damaged by unlikely large earthquake motion and experience consequent water leakage, the plant is able to resupply the needed make-up water to the pool to provide adequate cooling capability. Also, if the racks were to impact and experience some amount of permanent deformation, it is not likely that their primary function of keeping fuel assemblies in an upright position would be impaired. CR-3 is located in a low seismic activity zone. Thus, any significant ground motion associated with large dynamic excitation is not expected. Therefore, the staff concludes that the rack modules will perform their function during and after an SSE.

3.2.5 Summary

Based on the review of the licensee's submittal, additional information and analysis, as well as the staff's independent assessment, it is concluded that the design of the spent fuel rerack modules and the spent fuel pool is adequate to withstand the effects of the required operational and abnormal loads and is able to maintain integrity of the fuel assemblies and fuel rods. Furthermore, the design is in compliance with current licensing practice and, therefore, is acceptable.

Maintenance of uniform gaps between the racks and rack to the pool wall, as described in the licensee's submittal, is desirable from a structural point of view since it minimizes potential impact between the racks and the rack and pool wall. Therefore, in the event of an OBE, the licensee should inspect and maintain rack gaps. The surveillance should also include inspection of rack and fuel integrity for any damage.

3.3 Plant Systems

3.3.1 Coolant and Spent Fuel Temperature

3.3.1.1 Decay Heat Generation Rate

The licensee calculated the expected decay heat load when offloading a full core at the end of a 2-year cycle to be 33.0 MBTU/HR. The licensee stated that the expected heat load for all the previously offloaded spent fuel assemblies stored in the spent fuel pools is 0.5 MBTU/HR. This results in a total heat generation rate of 33.5 MBTU/HR. The licensee assumed infinite irradiation prior to refueling and a cooling period of 72 hours prior to removal. This calculation is conservative in that the guidelines for a full core offload allow assumption of a 150 hour cooling period. Therefore, the staff finds this methodology to be acceptably conservative.

3.3.1.2 Thermal Hydraulic Concerns

3.3.1.2.1 Spent Fuel Pool Coolant Temperatures

The licensee used the calculation of maximum decay heat generated during the full core (maximum abnormal case) offload to determine the temperature of the pool water leaving the spent fuel cooling system heat exchangers, and calculated it to be 157°F. This calculation assumes that both spent fuel pool cooling trains would be in operation. With only one train in operation, the pool temperature is estimated to be in excess of 210°F. However, the guidelines for this case, in SRP Section 9.1.3, specify that a single failure need not be considered. The calculated pool temperature of 157°F complies with the guideline temperature of less than 212°F and, therefore, is found to be acceptable.

The licensee did not calculate the case for the maximum normal offload for the reracked pool. However, since this case assumes one-third of the core is offloaded, the decay heat power level is approximately one-third that of the full core offload. The spent fuel cooling capacity is reduced by half since only one cooling train is assumed to be available. Therefore the resultant spent fuel coolant temperature will be less than in the full core offload case, 157°F.

The SRP guideline calls for a maximum temperature of 140°F in this case. However, the guideline specifies this temperature in order to protect demineralizer resins in spent fuel pool cleanup systems which may be affected by temperatures in excess of 140°F. The licensee assured the staff that the demineralizer resin used in the CR-3 cleanup system could withstand temperatures in excess of 250°F. Therefore, a maximum spent fuel pool temperature of 157°F is found to be acceptable in this case.

The licensee reported that the decay heat removal (DHR) system, which is permanently connected to the spent fuel pool cooling system, may be utilized to cool the spent fuel in the event of failure of both spent fuel cooling trains during a full core offload. Each DHR train is capable of producing a cooling effect of 30 MBTU/HR when used to cool the reactor coolant system. While both DHR trains may be needed, initially, to cool the spent fuel pool, one train should be capable of maintaining the spent fuel pool coolant temperature below 157°F as the rate of decay heat generation decreases. Additionally, the plant fire protection system could also be used to cool the spent fuel pool by the use of temporary connections and fire hoses.

In conclusion, the staff finds the spent fuel pool coolant temperatures under conditions of maximum normal refueling heat load and maximum abnormal (full core offloading) heat load and the use of other cooling systems in the event of failure of both spent fuel pool cooling system trains to be acceptable.

3.3.1.2.2 Maximum Spent Fuel Element Clad Temperatures

The licensee calculated the temperature of the fuel cladding assuming all spent fuel pool cooling methods were inoperable and that fuel pin cooling was established by natural circulation of the fluid in the pool.

The licensee assumed all downflow would occur at the sides of the spent fuel pool between the outermost storage racks and spent fuel pool walls with all bottom lateral flow between the bottom of the racks and bottom of the pool. The licensee iteratively solved the conservation equations of mass, momentum and energy to arrive at a solution. The licensee also examined cases of flow blockage up to 80%. The licensee reported that, in the worst case, "the fuel cladding temperatures are sufficiently low to preclude structural failures."

The staff finds the licensee's conclusion acceptable.

3.3.1.2.3 Maintenance of Spent Fuel Pool Coolant Level in the Event of Failure of Normal Cooling

The licensee noted that in the event of loss of one spent fuel cooling system train, the pool coolant temperature would rise to above 210°F, taking about 8 hours to increase from 160 to 190°F. If repairs could not be effected in that time period, an alternate source of water could be provided from the borated water storage tank at a rate in excess of that expected to boil off of spent fuel storage pool B once the pool coolant temperature reached the cooling point (greater than the approximate boil off rate of 70 gpm). In addition, the fire water system could be used to refill the pool at a rate of 500 gpm. The staff finds this to be acceptable.

3.3.2 Heavy Loads Handling Concerns

3.3.2.1 Load Drops on Spent Fuel

The licensee intends to move all the spent fuel in storage pool B into storage pool A before starting the reracking process. Therefore, movement of racks into and out of pool B cannot affect spent fuel. Thus, the heavy load drop concerns regarding radiation doses from gases escaping fuel and regarding criticality as a result of heavy load drops is not a concern in the reracking process. The movement of missile shields will not endanger any spent fuel by their removal and replacement. In addition, if dropped, they will float and, thus, do not constitute a heavy loads concern.

The remaining heavy loads issues are discussed below.

3.3.2.2 Spent Fuel Pool - Possible Unacceptable Damage as a Result of a Rack Drop

In the Safety Evaluation Report of July 5, 1974, the issue of a possible cask drop into an empty spent fuel pool was addressed. The staff then found the potential for a cask falling into the spent fuel storage pool (pool B) adjacent to the cask pool to be acceptable. This was based upon having the pool B empty and separated from the pool containing fuel, pool A, by a watertight gate (consisting of gate and inflatable seal) in place and sealed, as will be the case in the process of reracking spent fuel pool B. The Safety Evaluation Report of 1974 stated that the cask drop (in pool B) would not have an adverse effect on the fuel pool with the stored fuel assemblies, i.e. spent fuel storage pool A. It is noted, additionally, that the cask considered previously weighed 29 tons while the heaviest rack weighs 9 tons. Therefore, the staff considers the potential effects of a rack drop in pool B not to be a danger to spent fuel stored in pool A during the process of reracking and, thus, to be acceptable.

The licensee stated that the watertight gate had been stored on the wall between pool A and B, adjacent to its installation location. Thus, its employment should not endanger any spent fuel and is found to be acceptable.

3.3.2.3 Possible Damage to Safety-Related Equipment

The licensee reported that the reracking work would be performed by a contractor. The contractor is required to write procedures by which the reracking process will be conducted. The procedures will be reviewed by the Crystal River plant personnel to assure that they comply with the plant standards for movement of heavy loads throughout the plant. These include following safe load paths so as to prevent damage to redundant trains of safety-related safe shutdown equipment in the event of a heavy load drop. The staff finds the licensee's action to be acceptable.

3.3.2.4 Operator Training

The licensee committed to give a special course to the operators assigned by the contractor to handle the cranes used in the reracking process. This would assure the licensee that these operators were properly trained in moving heavy loads in compliance with applicable plant standards. The staff finds this to be acceptable.

3.3.2.5 Fixtures Used to Move Racks

The licensee stated that the contractor will, where applicable, design special fixtures to move racks in accordance with the criteria of ANSI 14.6-1976. Commercial standards will be used where the criteria of ANSI 14.6 are not required. The slings used by the contractor will be reviewed by the licensee to assure meeting applicable standards. The staff finds this to be acceptable.

3.3.2.6 Other Load Handling Concerns

Another load handling concern is that associated with criticality caused by dropping single fuel assemblies on fuel stored in the spent fuel pools during normal refueling. This is covered in a Section 3.1 above.

The staff considers the licensee's plan for reracking spent fuel pool B to be acceptable regarding spent fuel coolant temperatures, maximum cladding temperatures for spent fuel stored in the spent fuel pool, and regarding heavy load handling concerns related to damage to the spent fuel pool as a result of a rack drop and damage to safety-related equipment as a result of moving old racks out of and new racks into pool B.

3.4 Radiation Protection

3.4.1 Occupational Exposure Controls

The pool B rerack will be performed in accordance with CR-3 Administrative Instruction (AI)-1600, "ALARA Program Manual." All spent fuel assemblies have been moved to pool A. Thus, the major collective dose (3.2 person-rems) to plant personnel from the rerack operation will be due to removing, decontaminating, and shipping the spent fuel storage rack modules previously stored in pool B; and pool cleanup operations, including handling and processing of vacuum filters. Since CR-3 had already performed a similar rerack operation for pool A, the experience gained by plant personnel from this earlier operation will be helpful in the plant's ALARA planning. The SFP racks will be hydrolazed at site and the radiation levels will be reduced sufficiently to allow for exclusive use LSA shipments in strong, tight containers to be sent to the Westinghouse Waltz Mill Service Center near Madison, Pennsylvania, where the racks will be processed and prepared for disposal.

As part of the ALARA planning, the licensee plans to make use of remote handling tools as much as possible for jobs such as removing nuts from existing rack retainer bolts. To minimize possible contamination (e.g. hot particles) to personnel and plant facilities from the existing SFP racks after removal, high pressure water decontamination of these racks underwater will be conducted and strippable paint applied to the racks after removal from the pool. Portable lead shield blankets will then be used to minimize dose rates from these racks during removal to and storage on the 164 ft. elevation. The entire operation will be covered by the Plant Health Physicist and his staff. Radiation monitors will be placed around the work area and radiation surveys will be made to monitor the radiation levels around the SFP. All plant personnel working on this job will be covered by applicable Radiation Work Permits (RWP) with the appropriate protective clothing and, if needed, respiratory protective equipment; and personnel radiation monitoring equipment such as TLDs, pocket dosimeters, and again, if necessary, extremity dosimeters.

An isotopic analysis of the SFP water indicates that the primary radionuclides are Co-58, Co-60, Mn-54, Cs-134, and Cs-137 with concentrations of 1.16 E-5, 4.88 E-6, 1.06 E-6, 1.94 E-5, and 5.53 E-5 microcuries per cubic-centimeter respectively. These radionuclides are the primary source of radiation in the spent fuel pool. Operating experience with dense fuel storage in pool A has shown that dose rates of 0.5 to 2.0 mrem/hour are expected either at the edge or above the center of pool B after the rerack modification regardless of the quantity of fuel stored. This experience has also shown no noticeable increases in airborne radioactivity above the SFP and there has been no crud (e.g., Co-58, Co-60) build-up along the sides of the pool.

The major work effort to be expended for the rerack will be for the installation of the new racks. This is expected to be 2,560 man-hours excluding diving operations. To minimize dose to all personnel for this phase of the operation, pool B will be cleaned to a level in which the non-diving part of the installation operation will result in negligible dose to all personnel involved in this work. The use of divers will be minimized and the collective dose for this phase of the operation is expected to be 0.25 man-rem. Thus, the total collective dose projected for the entire pool B rerack modification is estimated to be 3.5 person-rems. This estimate is well below the historical range of collective doses for SFP reracking operations and is a small fraction of the approximately 262 person-rems per year that CR-3 has averaged over the past 3 years. Although the staff finds this estimate to be lower than expected, the staff believes CR-3's previous experience with reracking operations has improved the plant's capability to set and meet more stringent ALARA goals.

Based on our review of the CR-3 proposal, we conclude that the projected activities and estimated collective doses for this project appear achievable. The licensee intends to take ALARA considerations into account and to implement more stringent dose-reducing activities based on earlier operational experiences.

We conclude that the licensee will be able to maintain individual occupational radiation exposures within the limits of 10 CFR Part 20 and maintain doses ALARA, consistent with the guidelines of Regulatory Guide 8.8. Therefore, the proposed radiation protection aspects of the pool B rerack are acceptable.

3.4.2 Design Basis Accidents

In its application, the licensee evaluated the possible consequences of postulated design basis accidents (DBAs) and included means for their avoidance in the design and operation of the facility, and has provided means for mitigation of their consequences should they occur. The staff independently assessed these DBAs and agrees with the licensee that no new type of accident would be created by the installation and operation of the reracked spent fuel storage pool B.

In NRC's previous Safety Evaluation Report, dated July 9, 1974, the staff conservatively estimated offsite doses due to exposures to radionuclides released to the atmosphere from a fuel handling accident. This is the staff's scoping DBA for a spent fuel storage pool. The staff had concluded that the plant mitigative features would reduce the DBA doses to well below the doses specified in the applicable regulation, 10 CFR Part 100.

Since the licensee intends to increase the weight percent enrichment for its fuel assemblies in pool B from 4.0% to 4.2% weight percent of U-235 (pool A has already been authorized to store fuel assemblies up to 4.5%), which may allow fuel burnup greater than the current 33,000 megawatt days per metric ton (MWD/T), the staff reanalyzed the fuel handling DBA for this case. In a NRC contractor report by the Pacific Northwest Laboratory (PNL) entitled, "Assessment of the Use of Extended Burnup Fuel in Light Water Reactors," NUREG/CR 5009, February 1988, PNL examined the changes that could result in the NRC DBA assumptions, described in the various appropriate SRP sections and/or Regulatory Guides, that could result from the use of extended burnup fuel (up to 60,000 MWD/T). The staff agrees that the only DBA that could be affected by the use of extended burnup fuel, even in a minor way, would be the potential thyroid doses that could result from a fuel handling accident. PNL estimates that I-131 fuel gap activity in the peak fuel rod with 60,000 MWD/T burnup could be as high as 12%. This value is approximately 20% higher than the value normally used by the staff in evaluating fuel handling accidents (Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facilities for Boiling and Pressurized Water Reactors").

The staff, therefore, reevaluated the fuel handling accidents for the CR-3 facility with an increase in iodine gap activity in the fuel damaged in a fuel handling accident. Table 1 presents the fuel handling accident thyroid and whole body doses, without filters, presented in the operating licensing Safety Evaluation Report, dated July 9, 1974, and the increased thyroid doses (by 20%) resulting from extended burnup fuel.

TABLE 1

Radiological Consequences of Fuel
Handling Design Basis Accident (rem)

	Exclusion Area		Low Population Zone	
	Thyroid	Whole Body	Thyroid	Whole Body
Original Estimates (SER - 1974, Table 15.1)	57	<1	3	<1
Estimates for Higher Fuel Burnup*	68	<1	3.6	<1
Regulatory Requirement (10 CFR Part 100)	300	25	300	25

*Factor of 1.2 greater than original estimate for iodine

The staff concludes that the only potential increased doses resulting from a DBA with extended burnup to 60,000 MWD/T is the thyroid dose resulting from fuel handling accidents; these doses remain well within (25% or less, i.e. 75 rem) the 300 rem thyroid exposure guideline values set forth in 10 CFR Part 100. This small calculated increase is not significant.

3.5 Materials and Chemical Engineering

The stainless steel in the spent fuel pool liners and rack assemblies is compatible with the air saturated, borated water and radiation environment of the spent fuel pool. Borated water with 1925 ppm of boron would have approximately pH=5 and oxygen dissolved in water will help passivate the stainless steel. In this environment, austenitic stainless steel will exhibit only extremely low rates of corrosion. These corrosion rates are negligible for even the thinnest stainless steel elements of the pool liners or rack assemblies. Galvanic attack between the stainless steel in pool liners or rack assemblies and the Inconel/Zircaloy in the fuel assemblies will not be significant since these materials are protected by passivating oxide films. Boraflex is composed of non-metallic materials and, therefore, will not develop a galvanic potential with the metal components.

The vent holes will allow any gas that may be generated from the silicone polymer binder in the Boraflex due to heat and irradiation to escape, thus preventing possible bulging or swelling. Boraflex has undergone extensive testing to determine the effects of gamma irradiation in various environments and to verify its structural integrity and credibility as neutron absorbing material (Bisco Products, Inc., Technical Report No. NS-1-001, "Irradiation Study of Boraflex Neutron Shielding Materials," August 12, 1981). The evaluation tests have shown that Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. Tests performed at the University of Michigan consisted of exposing Boraflex in 2000 ppm boron solution to 1.03E11 rads of gamma radiation with a concurrent neutron flux of 8.3E13 neutrons/sq cm/sec. These tests indicate that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma and neutron irradiation. However, irradiation caused some loss of flexibility and shrinkage of the Boraflex. Long-term water soak tests at high temperatures were also conducted. They are described in "Boraflex Neutron Shielding Product Performance Data," August 25, 1981. The tests show that Boraflex will withstand a temperature of 240°F in a solution of 3000 ppm boron for 251 days without visible distortion or softening. The Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide. The spent fuel pool water temperature is normally maintained below 140°F which is well below the 240°F test temperature.

The tests referenced above have shown that neither irradiation, environment, nor Boraflex composition have a discernible effect on the neutron transmission of the Boraflex material. The tests also have shown that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions are reached regarding the leaching of elemental boron from the Boraflex. Boron carbide in the Boraflex typically contains 0.1 weight percent of soluble boron. The test results have confirmed the encapsulation capability of silicone polymer matrix to prevent the leaching of soluble species from the boron carbide.

Anomalies ranging from minor physical changes in color, size, hardness and brittleness to formation of gaps up to four inches in width were observed in Boraflex panels that have been used in the spent fuel pools of four nuclear plants. The exact mechanism that causes the observed physical degradation of Boraflex has not been confirmed. The staff postulates that gamma radiation from the spent fuel may induce crosslinking of the polymer in the Boraflex, producing shrinkage of the Boraflex material. When crosslinking becomes saturated, scissioning (a process in which bonds between atoms are broken) of polymer predominates as the accumulated radiation dose increases. Scissioning may produce porosity, which allows spent fuel pool water to permeate the Boraflex material and may cause embrittlement. Gamma radiation from the spent fuel is the most probable cause of the observed physical degradation, such as color change, size, hardness and brittleness. The staff does not have sufficient information to determine conclusively what caused the gap formation in some Boraflex panels. However, it is conceivable that if both ends of a full-length Boraflex panel are physically restrained, then shrinkage caused by gamma radiation may promote panel tearing and subsequent gap formation.

The staff has determined that reasonable assurance exists that the Boraflex panels are not physically restrained in the design of the storage racks at CR-3. The wrappers hold the Boraflex in place on the sides of the cells without pinching, binding, sagging or buckling. Therefore, it is not likely that gaps will develop to any significant extent in the Boraflex panels during the design life of spent fuel storage racks. However, minor physical degradation may take place due to irradiation of the Boraflex panels. In the unlikely event of gap formation in the Boraflex panels that would lead to loss of neutron absorbing capability, the monitoring program would detect such degraded Boraflex panels and the licensee would have sufficient time to perform a criticality evaluation and to take the appropriate corrective action.

Summary

Based on the above discussion, the staff concludes that corrosion of the proposed fuel storage racks due to the spent fuel pool environment should be of little significance during the life of the facility. The surveillance program proposed by the licensee would reveal any deterioration in neutron absorbing capability of Boraflex and, if a significant degradation is found, the licensee would have sufficient time to take the appropriate corrective action.

The staff finds that the selection of appropriate materials of construction and development of the proposed Boraflex surveillance program meet the requirements of 10 CFR Part 50 Appendix A, General Design Criterion 61, regarding the capability to permit appropriate periodic inspection and testing of components and General Design Criterion 62 regarding prevention of criticality by the use of neutron absorbers and by maintaining structural integrity of components and are, therefore, acceptable.

4.0 STATE CONSULTATION

Based upon the written notice of the proposed amendment, the Florida State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32 and 51,35, environmental assessments and findings of no significant impacts were prepared and published in the Federal Register (55 FR 35479 and 54 FR 35954). Accordingly, based upon the environmental assessment, the NRC staff has determined that the issuance of this amendment will not have significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: L. Kopp
S. B. Kim
N. Wagner
J. Wang
K. Parczewski
H. Silver

Date: April 16, 1991

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

FLORIDA POWER CORPORATION

DOCKET NO. 50-302

NOTICE OF WITHDRAWAL OF PORTION OF AMENDMENT APPLICATION TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has granted a request by Florida Power Corporation (the licensee) to withdraw a portion of its October 31, 1989 application for an amendment to Facility Operating License No. DRP-72, issued to the licensee for operation of the Crystal River Unit 3 Nuclear Generating Plant, located in Citrus County, Florida. Notice of Consideration of Issuance of this amendment was published in the FEDERAL REGISTER on March 12, 1990 (55 FR 9230).

The purpose of the licensee's amendment request was to revise the Technical Specifications (TS) to increase the capacity of spent fuel storage pool B and increase the allowable enrichment in fuel pool B.

Subsequently the licensee informed the staff that the portion of the amendment application which requested a one-time relief to allow removal of the missile shields over spent fuel pool B while modifying the pool racks is no longer required. Thus, this portion of the amendment application is considered to be withdrawn by the licensee.

For further details with respect to this action, see (1) the application for amendment dated October 31, 1989, as supplemented January 25, March 8, June 21, August 23, November 8, and November 28, 1990, and (2) Amendment No. 134 dated April 16, 1991 .

These documents are available for public inspection at the Commission's

Public Document Room, the Gelman Building, 2120 L Street, N.W., Washington, D.C. and at the Coastal Region Library, 8619 W. Crystal Street, Crystal River, Florida 32629.

Dated at Rockville, Maryland, this 16thday of April 1991.

FOR THE NUCLEAR REGULATORY COMMISSION

Harley Silver, Project Manager
Project Directorate II-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation