October 25, 2007

Mr. Dale E. Young, Vice President
Crystal River Nuclear Plant (NA1B)
ATTN: Supervisor, Licensing & Regulatory Programs
15760 W. Power Line Street
Crystal River, Florida 34428-6708

SUBJECT: CRYSTAL RIVER, UNIT 3 - ISSUANCE OF AMENDMENT REGARDING FUEL STORAGE PATTERNS IN THE SPENT FUEL POOL (TAC NO. MD3308)

Dear Mr. Young:

The Commission has issued the enclosed Amendment No. 227 to Facility Operating License No. DPR-72 for Crystal River Unit 3 in response to your letter dated October 5, 2006, as supplemented by letters dated April 4 and July 19, 2007. The amendment changes the restrictions on fuel storage in the spent fuel pool.

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

Sincerely,

/RA/

Stewart N. Bailey, Senior Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-302

Enclosures:
1. Amendment No. 227 to DPR-72
2. Safety Evaluation

cc w/enclosures: See next page

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Docket No. 50-302

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OFFICIAL RECORD ***w/ comments**

FLORIDA POWER CORPORATION

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CITY OF NEW SMYRNA BEACH

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ORLANDO UTILITIES COMMISSION AND CITY OF ORLANDO

SEMINOLE ELECTRIC COOPERATIVE, INC.

DOCKET NO. 50-302

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.227
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 5, 2006, as supplemented by letters dated April 4 and July 19, 2007, 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. 227, 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 and shall be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA by Evangelos C. Marinos/

Thomas H. Boyce, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Operating License
and Technical Specifications

Date of Issuance: October 25, 2007

ATTACHMENT TO LICENSE AMENDMENT NO. 227

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

Replace the following page of Facility Operating License DPR-72 with the attached revised page. The revised page is identified by amendment number and contains a vertical line indicating the area of change.

Remove	Insert
4	4

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

Remove	Insert
3.7-32	3.7-32
3.7-33	3.7-33

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 227 TO FACILITY OPERATING LICENSE NO. DPR-72

FLORIDA POWER CORPORATION, ET AL.

CRYSTAL RIVER UNIT 3 NUCLEAR GENERATING PLANT

DOCKET NO. 50-302

1.0 INTRODUCTION

By letter dated October 5, 2006, as supplemented by letters dated April 4 and July 19, 2007, the Florida Power Corporation (the licensee) requested changes to the Technical Specifications (TSs) for Crystal River Unit 3 (CR-3). The proposed changes would alter the restrictions on fuel loading in the CR-3 spent fuel pools (SFPs). To support the change, the licensee submitted a new criticality analysis, performed by Holtec International, for the fuel storage racks.

The supplements dated April 4 and July 19, 2007, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on November 21, 2006 (71 FR 67394).

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Appendix A, Criterion 62, "Prevention of criticality in fuel storage and handling," states, "Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations."

As stated in 10 CFR 50.68(b)(1), "Plant procedures shall prohibit the handling and storage at any one time of more fuel assemblies than have been determined to be safely subcritical under the most adverse moderation conditions feasible by unborated water."

As stated in 10 CFR 50.68(b)(4), "If no credit for soluble boron is taken, the k-effective [k_{eff}, the effective neutron multiplication factor] of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with unborated water. If credit is taken for soluble boron, the k-effective of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95 percent probability, 95 percent confidence level, if flooded with borated water, and the k-effective must remain below 1.0 (subcritical), at a 95 percent probability, 95 percent confidence level, if flooded with unborated water."

As stated in 10 CFR 50.36(d)(4), "Design features to be included are those features of the facility such as materials of construction and geometric arrangements, which, if altered or modified, would have a significant effect on safety and are not covered in categories described in paragraphs (d) (1), (2), and (3) of this section."

The revised CR-3 SFP criticality analysis takes credit for soluble boron during accident scenarios. CR-3 TS 4.3, "Fuel Storage," requires the k_{eff} to be less than or equal to 0.95 when flooded with unborated water. The licensee's analysis used an acceptance criterion of k_{eff} less than or equal to 0.95 for the case with unborated water, which is consistent with TS 4.3. For the accident scenarios that assume the SFP is flooded with borated water, the revised CR-3 SFP criticality analysis uses an acceptance criterion of k_{eff} less than or equal to 0.9450, which provides margin to the regulatory requirement. However, the revised criticality analysis does not include all of the soluble boron that would reasonably be in the SFP, thereby providing additional margin to the regulatory requirement.

3.0 <u>PROPOSED CHANGE</u>

The CR-3 SFPs are described in Section 9.6.1.2, "Fuel Storage," of the CR-3 Final Safety Analysis Report (FSAR). CR-3 has two SFPs that are designated Pool A and Pool B. The Pool A racks are a different design than the Pool B racks, so the fuel storage limitations are different for the two pools. The storage requirements are specified in TS 3.7.15, "Spent Fuel Assembly Storage."

For Pool A, TS 3.7.15 currently divides the fuel into two categories based on burnup and initial enrichment. The highest reactivity (e.g., highest enrichment/lowest burnup) category must be stored in a one out of two checkerboard arrangement with empty cells, which requires all face adjacent cells to this category be empty. The other category may be stored without restriction, other than it cannot cause a violation of the restriction on the highest reactivity category. The proposed TS 3.7.15 will divide the Pool A fuel into three categories. For the highest reactivity category, Category F, the burnup/enrichment criteria are not changed from the current requirements. The lower reactivity category is divided in two categories with Category B being the least reactive fuel and the remainder being Category A. With this new division, the highest reactive fuel, Category F, must be stored in a checkerboard arrangement with empty cells or the lowest reactive fuel, Category B, and Category A fuel must be separated from the checkerboard by a row of Category B fuel. This will increase the Pool A storage capacity by allowing Category B fuel to be stored in cells that were previously required to be empty.
TS Figure 3.7.15-1 is being revised to include the new category designations, the curve differentiating Category B from Category A, and footnotes describing the limitations on the categories.

For Pool B, TS 3.7.15 currently divides the fuel into three categories. The highest reactivity category was prohibited from being stored in Pool B. The criteria for establishing this category are not being changed from current requirements, but the category is being named Category BE. Under the revised TS 3.7.15, Category BE fuel may be stored in Pool B provided it is surrounded by eight empty water cells and these empty cells may not be shared with another Category BE fuel assembly. Currently, the medium reactivity category may only be stored in periphery cells, which are those cells with at least one side (face) adjacent to the SFP wall. The criteria for establishing which fuel is in this category are not being changed from the current requirements, but the category is being named Category BP. Under the revised TS 3.7.15, Category BP fuel can still only be stored in periphery cells; however, the definition of a periphery cell is being changed. Under the new definition, a cell may be considered a periphery cell if it has at least one side adjacent to an empty cell that, in turn, has at least one side face adjacent to the SFP wall. A Category BP fuel assembly may be stored in this “periphery” cell provided it does not cause a violation of Category BE fuel requirements. Currently the lowest reactivity category may be stored without restriction. The criteria for establishing which fuel is in this category are not being changed from the current requirements, but the category is being named Category B. The criteria for designating fuel Category B are the same for both pools. Under the revised TS 3.7.15, Category B fuel can still be stored without restriction as long as it does not cause a violation of the Category BE or BP requirements. TS Figure 3.7.15-2 is being revised to include the new category designations and footnotes describing the limitations on the categories.

4.0 TECHNICAL EVALUATION

The Pool A storage cells are composed of stainless steel walls sandwiching a Carborundum neutron absorber panel, centered on each side of the storage cell. The steel walls define the storage cells and the flux-trap water gap used to augment reactivity control. The cells are located on a pitch of 10.50 inches, with a flux trap water gap of 1.227 inches. Additional details are provided in Table 5.7 and Figure 5.2 of Reference 1. The Pool A racks also contain neutron absorber on the peripheral faces of the rack modules, facing either another rack module or the SFP wall.

As described above, the Pool A fuel assemblies will be placed into one of three categories based on combined enrichment and burnup criteria. The most reactive fuel, Category F, must be checkerboarded with the least reactive fuel, Category B, or empty cells. Category A and B fuel may be stored together without restriction. Category A fuel assemblies must be separated from the checkerboard by at least one row of Category B fuel.

The Pool B storage cells are composed of a single Boral neutron absorber panel affixed to each stainless steel wall of a storage cell. Stainless steel sheathing is used to affix the neutron absorber to the storage cell walls, and to position the neutron absorber in the center of the cell. The cells are located on a pitch of 9.11 inches, with no flux trap water gap. Additional details are provided in Table 5.8 and Figure 5.4 of Reference 1. There is no degradation assumed for the Boral. The Boral neutron absorber in the Pool B racks is 144 inches in length, which is approximately the same length as the active fuel region of the Babcock and Wilcox (B&W) 15x15 fuel assemblies. The licensee stated that potential misalignment of the Boral is considered.

As described above, the Pool B fuel assemblies will be placed into one of three categories based on combined enrichment and burnup criteria. The most reactive fuel, Category BE, must be surrounded by eight empty cells, and those empty cells may not be shared with another Category BE fuel assembly. The Category BP assemblies must be placed in periphery cells, using the new definition of periphery. Category B fuel may be stored without restriction as long as it does not violate the requirements placed on Category BE or BP fuel.

4.1 Computer Codes

The criticality analysis uses two computer codes, MCNP4a and CASMO-4, Version 2.05.14. MCNP4a is a continuous energy three dimensional Monte Carlo code developed at the

Los Alamos National Laboratory. CASMO-4 is a two-dimensional multigroup transport theory code based on capture probabilities.

The principal method for the criticality analysis of the high density storage racks is MCNP4a. The licensee selected MCNP4a because it has been extensively verified and used for criticality analyses, and it has all of the necessary features for this analysis. For this analysis, MCNP4a calculations use continuous energy cross-section data based on ENDF/B-V and ENDF/B-VI, with the exception of two lumped fission products calculated by the CASMO-4 depletion code that do not have corresponding cross sections in MCNP4a. For these isotopes, the CASMO-4 cross sections are used in MCNP4a. Holtec has performed analyses to validate this modeling assumption, showing that the cross sections result in the same reactivity effect in both CASMO-4 and MCNP4a. Benchmark calculations show a bias in k_{eff} of 0.0009 and an uncertainly of plus or minus 0.0011 evaluated with a 95-percent probability, 95-percent confidence level.

For the fuel depletion analysis, CASMO-4 was used to determine the isotopic composition of the spent fuel, using a 70-group cross-section library. In addition, the CASMO-4 was used to determine the reactivity effect of fuel and rack tolerances, temperature variation, depletion uncertainty, and to perform other studies. This was done by restarting in the rack geometry, which yields the two-dimensional infinite neutron multiplication factor for the storage rack.

This approach is identical to the evaluation that Holtec performed for Arkansas Nuclear One (ANO) Unit 1 (Reference 4), which was accepted by the NRC in Reference 5. Both ANO Unit 1 and CR-3 use B&W 15x15 fuel assembly designs. The SFP racks at both plants are similar with respect to materials, rack dimensions, and the use of permanently installed neutron absorbers. Therefore, this use of computer codes is acceptable based on the ANO Unit 1 precedent.

4.2 Biases and Uncertainties

Fuel Assembly Mechanical Uncertainties

The criticality analysis uses two assembly designs, the B&W Mark B-10F and the Mark B-11. However, the licensee identified the following additional fuel assembly designs that are either in use or have previously been used at CR-3: Mark B-10, Mark B-9, Mark B-4, Mark B-3, and Mark B/HTP. The licensee stated the Mark B-4 and Mark B-3 are identical to Mark B-10 and Mark B-9, except for fuel density and pellet diameter, for which Mark B-10 and Mark B-9 are bounding. In response to staff's request for additional information (RAI), the licensee provided information that compared the k_{eff} for Mark B-10F, Mark B-10, Mark B-9, Mark B-11, and Mark B/HTP fuel assemblies for burnup ranging from 0 through 45 gigawatt days per metric ton of Uranium (GWD/MTU). This information shows the Mark B-10F and B-11 to be the limiting assemblies, making them appropriate for use in the criticality analysis.

The licensee's analysis determined separate uncertainties for the fuel designs for each pool. The licensee evaluated the manufacturing tolerances of the bounding assemblies by conducting sensitivity studies to determine the most reactive condition of those tolerances. The licensee performed the studies at different enrichment and burnup combinations. The limiting uncertainties were applied to the analysis. The licensee provided examples of the results of these analyses showing that k_{eff} was below 0.95 with bias and uncertainties included.

Spent Fuel Pool Storage Rack Uncertainties

The licensee's analysis determined separate uncertainties for each pool. The licensee evaluated the manufacturing tolerances of the storage racks by conducting sensitivity studies to determine the most reactive condition of those tolerances. The licensee performed the studies at different enrichment and burnup combinations. The limiting uncertainties were applied to the analysis. The licensee provided examples of the results of these analyses showing that k_{eff} was below 0.95 with bias and uncertainties included.

Burnup Uncertainty

The licensee's analysis assumed an uncertainty of 5 percent of the total reactivity decrement associated with the credited burnup. This is consistent with the established staff position, as documented in Reference 3, for bounding the effects of uncertainties in these analyses and is, therefore, acceptable.

Temperature Bias

The licensee's analysis determined that both SFP racks had a negative moderator temperature coefficient. Therefore, the most reactive moderator conditions would be full density water, which occurs at 4° C. In MCNP4a, the Doppler treatment and cross sections are valid only at 300K (~27° C). Therefore, the analysis applied a temperature bias. A temperature bias was determined from an analysis of an infinite array of the bounding fuel assemblies for both fresh and depleted fuel assemblies. The largest temperature bias was used in the analysis.

4.3 Fuel Characterization

For the SFP criticality analysis, the fuel must be characterized appropriately. Characterization of fresh fuel is relatively straight forward. It is based primarily on Uranium 235 (^{235}U) enrichment and various manufacturing tolerances. The manufacturing tolerances are typically manifested as uncertainties, as discussed above, or are bounded by values used in the analysis, and these tolerances and bounding values would also carry through to the spent fuel. However, the characterization of spent fuel is more problematic. In addition to the manufacturing tolerances, its characterization is based on the specifics of its initial conditions and its operational history in the reactor.

NUREG/CR 6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR [light water reactor] Fuel" (Reference 6), provides a discussion of the treatment of depletion analysis parameters. While NUREG/CR 6665 is focused on criticality analysis in storage and transportation casks, the basic principals with respect to the depletion analysis apply generically to SFPs, especially when the discussion includes the effect in an infinite lattice criticality analysis, similar to the analysis performed for SFPs. The basic premise is to select parameters that maximize the Doppler broadening and spectral hardening of the neutron field, resulting in maximum Plutonium 241 (^{241}Pu) production. NUREG/CR 6665 discusses six parameters affecting the depletion analysis: fuel temperature, moderator temperature, soluble boron, specific power and operating history, fixed burnable poisons, and integral burnable poisons.

For fuel and moderator temperatures, NUREG/CR 6665 recommends using the maximum operating temperatures to maximize ^{241}Pu production. NUREG/CR 6665 approximates the moderator temperature effect for an infinite lattice of high burnup fuel to be 90 percent mil per degree Fahrenheit (pcm/°F). Thus a 10° F change in moderator temperature used in the depletion analysis would result in a 0.005 change in k_{eff}. The licensee's analysis used a moderator temperature of 604° F, which was described as the core average moderator temperature at the top of the active region. In response to staff's RAI, the licensee stated that 604° F is the highest temperature that any portion of the active fuel region would experience during reactor operation. Applying that temperature to the entire length of the active fuel region is conservative. The licensee's analysis used a fuel temperature of 1238° F, which was described as the core average fuel temperature. In response to the staff's RAI, the licensee identified 1146° F as the maximum average fuel temperature. Section 3.2.3.2.3.g.4 of Revision 30 of the CR-3 FSAR indicates the equilibrium cycle end of life (EOL) fuel average temperature is 1250° F. Since EOL fuel temperatures are typically higher than beginning of life, the EOL temperature would be a better indication of the maximum average fuel temperature. The 1238° F used is less than the EOL fuel average temperature stated in the CR-3 FSAR, but is acceptable for this analysis when taken in context with the other neutron spectrum hardening parameters.

For boron concentration, NUREG/CR 6665 recommends using a conservative cycle average boron concentration. The licensee's analysis used a boron concentration of 1000 parts per million (ppm) throughout the depletion of the fuel assemblies. A review of Chapter 3 of the CR-3 FSAR indicates that 1000 ppm is conservative with respect to a cycle-average soluble boron concentration. Therefore, the staff finds it acceptable.

For power and operating history, NUREG/CR 6665 does not have a specific recommendation. NUREG/CR 6665 estimated this effect to be on the order of a 0.002 change in k_{eff} using the operating histories it considered. Based on the difficulty of reproducing a bounding or even a representative power operating history, NUREG/CR 6665 merely recommends using a constant power level and retaining sufficient margin to cover the potential effect of a more limiting power history. The licensee used a constant core power for the depletion calculations and appears to have ample margin.

For fixed burnable poisons such as burnable poison rod assemblies (BPRAs) and axial power shaping rod assemblies (APSRs), NUREG/CR 6665 recommends that the depletion calculation include the maximum loading. BPRAs have more reactivity worth than APSRs. The Holtec analysis starts with a BPRA in the fuel assemblies, and then transitions to an APSR when the BPRA is depleted to the point where the APSR has more reactivity worth. In response to the staff's RAI regarding the tolerances associated with BPRAs and APSRs, the licensee indicated that modeling every assembly with BPRA followed by an APSR contained sufficient conservatism to bound any effect the tolerances for BPRAs and APSRs may have. The staff agrees and finds the licensee's treatment of the BPRAs and APSRs acceptable.

NUREG/CR 6665 does not have a specific recommendation for integral burnable poisons. The licensee's analysis does not model integral burnable poisons; therefore, it does not take credit for integral burnable poisons or their effect on neutron spectrum hardening. The NRC staff believes the licensee's modeling of all assemblies with a BPRA followed by a APSR provides sufficient margin with respect to spectral hardening effects to compensate for the effect of integral burnable poisons.

Another important aspect of fuel characterization is the selection of the burnup profile. At the beginning of life, a pressurized-water reactor (PWR) fuel assembly will be exposed to a near-cosine axial-shaped flux, which will deplete fuel near the axial center at a greater rate than at the ends. As the reactor continues to operate, the cosine flux shape will flatten because of the fuel depletion and fission-product buildup that occurs near the center. Near the fuel assembly ends, burnup is suppressed due to neutron leakage. If a uniform axial burnup profile is assumed, then the burnup at the ends is over predicted. Analysis has shown that this results in an under prediction of k_{eff}, and generally the under prediction becomes larger as burnup increases. This is known as the end effect. Judicious selection of the axial burnup profile is necessary to ensure k_{eff} is not underpredicted due to the end effect. NUREG/CR 6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis” (Reference 7), provides insight for selecting an appropriate axial burnup profile.

The licensee conducted a site-specific analysis to determine the appropriate burnup axial profiles. For fuel assemblies with axial blankets, the licensee’s analysis determined a flat axial burnup profile was appropriate. This is consistent with NUREG/CR 6801, which indicates the lower enrichment in the axial blankets acts to balance the mechanism which causes the end effect. For fuel assemblies without axial blankets the licensee’s analysis used a single 10-zone axial burnup profile for the entire burnup range. This is counter to NUREG/CR 6801, which recommends using additional zones and burnup-dependent axial profiles. NUREG/CR 6801 identifies bounding axial profiles for 12 burnup groups, but indicates this can be reduced to three groups if the appropriate bounding profiles are used. The three burnup groups are 0-18 GWD/MTU, 18-30 GWD/MTU, and greater than 30 GWD/MTU. The profile the licensee used is similar to the profile recommended for burnup greater than 30 GWD/MTU. The primary difference is that the licensee used larger zones in the center, while using similarly sized zones on the end. There is excellent agreement between the licensee’s profile and the profile recommended by NUREG/CR 6801; therefore, the staff concludes that the licensee’s burnup profile is acceptable for evaluating burnup greater than 30 GWD/MTU. However, Figure 3 of NUREG/CR 6801 shows that using the profile for burnup greater than 30 GWD/MTU on fuel with lower burnup will underpredict k_{eff}. Since the limiting profiles in NUREG/CR 6801 are primarily those from B&W 15x15 fuel assemblies, it is reasonable to expect them to be applicable to CR-3. The licensee’s site specific analysis considers the checkerboarded fresh and depleted assemblies, which differs from the specific case analyzed in NUREG/CR 6801. The fresh assemblies do not have an axial profile and this, coupled with their higher reactivity worth, will ameliorate the end effect associated with the depleted assemblies. In Table 7.7 of Reference 1, the licensee provided sample results from analysis performed to determine the most reactive profile. That information indicates the flat profile produces the most reactive scenario for this specific analysis.

Therefore, the staff concludes that the licensee has adequately demonstrated that k_{eff} will remain below 0.95 for the unborated conditions in Pool A and Pool B for the storage patterns in proposed TS 3.7.15, as required by TS 4.3.1.1.b.

4.4 Determination of Soluble Boron Requirements

The regulatory requirement for soluble boron credit is that the k_{eff} of the spent fuel storage racks loaded with fuel of the maximum fuel assembly reactivity must not exceed 0.95, at a 95-percent probability, 95-percent confidence level, if flooded with borated water. The licensee’s analysis used an acceptance criterion of k_{eff} less than 0.9450 at a 95-percent probability, 95-percent confidence level.

To determine the soluble boron requirement for Pool A, the licensee's analysis considered a checkerboard array of Category B and Category F assemblies with one fresh assembly at 5.0 weight percent (w/o) ^{235}U enrichment misloaded in the place of a Category B assembly. Thus, the misloaded assembly has another 5.0 w/o fresh assembly adjacent to each face and a Category B assembly on each corner. The misloaded assembly is modeled at the center of a 5x5 array with periodic boundary conditions. This is reasonably the most reactive configuration for a misloading of a single assembly. In response to staff's RAI, the licensee provided the results of the cases it considered for various enrichment and burnup combinations for Category B assemblies. Those results show that the initial enrichment of 4.5 w/o with 37.46 GWD/MTU of burnup required the most soluble boron. This scenario required 165 ppm of soluble boron to meet the licensee's acceptance criterion of maintaining k_{eff} less than 0.9450 at a 95-percent probability, 95-percent confidence level, including uncertainties.

To determine the soluble boron requirement for Pool B, the licensee's analysis considered a 5.0 w/o fresh assembly misloaded in a cell that is face adjacent to the Category BE fuel assembly as the most reactive scenario. This scenario is modeled in a 7x7 array with periodic boundary conditions. The misloaded assembly would be face adjacent on one side with a 5.0 w/o fresh Category BE assembly, and face adjacent on the opposite side with a Category B assembly. The remaining two faces would be adjacent to empty cells. Two corners of the misloaded assembly would be adjacent to Category B fuel assemblies and two corners would be adjacent to empty cells. This is reasonably the most reactive scenario of the changes to Pool B storage. In response to the staff's RAI, the licensee provided the results of the cases it considered for various enrichment and burnup combinations for Category B assemblies. Those results show that the initial enrichment of 3.5 w/o with 25.47 GWD/MTU of burnup required the most soluble boron. This scenario required 46 ppm of soluble boron to meet the licensee's acceptance criterion of maintaining k_{eff} less than 0.9450 at a 95-percent probability, 95-percent confidence level, including uncertainties.

CR-3 TS 3.7.14, "Spent Fuel Pool Boron Concentration," requires the SFP soluble boron concentration to be 1925 ppm. This is a factor of 10 higher than the boron concentrations determined by the licensee's analyses. Therefore, the staff concludes that the licensee has adequately demonstrated that the TS 3.7.14 soluble boron requirement is sufficient to support the TS 4.3.1.1.b criticality requirement.

5.0 STATE CONSULTATION

Based upon a letter dated May 2, 2003, from Michael N. Stephens of the Florida Department of Health, Bureau of Radiation Control, to Brenda L. Mozafari, Senior Project Manager, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

6.0 ENVIRONMENTAL CONSIDERATIONS

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (71 FR 67394). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

7.0 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

8.0 REFERENCES

1. Progress Energy letter 3F1006 01 from Daniel L. Roderick, Director Site Operations Crystal River Nuclear Plant, to NRC document control desk, "Crystal River Unit 3 License Amendment Request # 292, Revision 0, Additional Storage Patterns for Crystal River Unit 3 Storage Pools A and B," October 5, 2006. (Agencywide Documentation and Management System (ADAMS) Accession No. ML062830073)

2. Progress Energy letter 3F0707 05 from Dale E. Young, Vice President, Crystal River Nuclear Plant, to NRC document control desk, "Crystal River Unit 3 License Amendment Request # 292, Revision 1, Additional Storage Patterns for Crystal River Unit 3 Storage Pools A and B and Response to Request for Additional Information (TAC No. MD3308)," July 19, 2007. (ADAMS Accession No. ML072040278)

3. NRC Memorandum from L. Kopp to T. Collins, "Guidance on the Regulatory Requirements for Criticality Analysis of Fuel Storage at Light Water Reactor Power Plants," August 19, 1998. (ADAMS Accession No. ML003728001)
4. Entergy Operations, Inc., letter 1CAN070603, 3F0707 05 from Jeffery S. Forbes, Vice President, Operations ANO, to NRC document control desk, "License Amendment Request to Support the Use of Metamic7 Poison Insert Assemblies in the Spent Fuel Pool Arkansas Nuclear One, Unit 1, Docket No. 50 313, License No. DPR 51," July 27, 2006. (ADAMS Accession No. ML062220440)

5. NRC Letter Mr. Jeffrey S. Forbes, Site Vice President, Arkansas Nuclear One, "Arkansas Nuclear One, Unit No. 1 - Issuance of Amendment for use of METAMIC7 Poison Insert Assemblies in the Spent Fuel Pool (TAC No. MD2674)," dated January 26, 2007, (ADAMS Accession No. ML070160038)

6. NUREG/CR 6665, "Review and Prioritization of Technical Issues Related to Burnup Credit for LWR Fuel," (ADAMS Accession No. ML003688150)

7. NUREG/CR 6801, "Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analysis." (ADAMS Accession No. ML03110292)

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