Florida Power
CORPORATION
Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72

October 30, 1998
3F1098-15

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555-0001

Subject: License Amendment Request #245, Revision 0
Revision to Licensing Basis - Methodology Change for Boraflex Degradation

Reference: NRC to FPC letter, 3N1295-08, dated December 15, 1995, "Issuance of Amendment Re: Fuel Enrichment Increase (TAC No. M91536)"

Dear Sir:

Florida Power Corporation (FPC) hereby submits a request for an amendment to the licensing basis for Crystal River Unit 3 (CR-3), Facility Operating License No. DPR-72. The attached License Amendment Request (LAR) #245, Revision 0, proposes a change to the methodology for the Spent Fuel (SF) Pool B criticality analysis. This proposed change is necessary due to Boraflex degradation in the SF Pool B at CR-3. FPC has concluded that the change in the criticality analysis methodology poses no significant hazards. However, it does represent an unreviewed safety question (USQ) in that utilizing the current NRC-approved calculational methodologies and assumptions for CR-3, the Boraflex degradation would result in a reduction of the margin of safety. The new analysis uses current regulatory accepted calculational methodologies based on a realistic set of assumptions.

The current licensing basis was approved by the NRC as part of the above referenced License Amendment. Recent Boraflex samples from the SF Pool B demonstrate a weight loss in excess of the available margin within the current licensing basis calculation. The criticality analysis calculations proposed by this LAR demonstrate that the burnup/enrichment curves in the current Improved Technical Specifications (ITS) have sufficient margin to accommodate up to a 20% loss in Boraflex neutron absorption, and still maintain SF Pool B at less than or equal to 0.95 k-effective when fully loaded and flooded with unborated water. Therefore, using the proposed analysis, the design basis and safety function are maintained in accordance with the requirements of ITS Limiting Condition for Operation (LCO) 3.7.15. Similar methodology to that being proposed in this LAR has been utilized in part and licensed for SF Pool A at CR-3, at Waterford Unit 3, and at Three Mile Island Unit 1.

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An evaluation of the proposed change to the licensing basis is provided in Attachment A. The proposed licensing basis changes to the Final Safety Analysis Report (FSAR), and associated ITS Bases, in strikeout/highlight format, reflect the means to document the approval of the methodology change and are provided in Attachment B. The report summarizing the new analysis from Holtec International, HI-982056, is provided in Attachment C.

Based on the guidance provided in NRC Generic Letter 91-18, Revision 1, FPC has determined that the Boraflex degradation represents a nonconforming condition, but the SF Pool B remains operable. Deficiency Report (DR) 98-3327 was developed to document this decision, and is available for NRC review at the facility.

FPC is requesting approval of this change to the CR-3 licensing basis by mid-June 1999, to support receipt of new fuel. This letter establishes no new regulatory commitments. If you have any questions regarding this letter, please contact Ms. Sherry Bernhoft, Manager, Nuclear Licensing at (352) 563-4566.

Sincerely,

MW Rencheck

M.W. Rencheck
Director
Nuclear Engineering and Projects

MWR/lrm

xc: Regional Administrator, Region II
Senior Resident Inspector
NRR Project Manager

Attachments:

A. License Amendment Request, No Significant Hazards Consideration Evaluation, and Environmental Impact Evaluation
B. Proposed Revisions to Final Safety Analysis Report and Improved Technical Specification Bases in Strikeout/Highlighted Format
C. Holtec Report, HI-982056, "Reanalysis of the Crystal River Pool B Spent Fuel Storage Racks with Boraflex Degradation"

STATE OF FLORIDA

COUNTY OF CITRUS

Michael W. Rencheck states that he is the Director, Nuclear Engineering and Projects for Florida Power Corporation; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

Michael W. Rencheck

Michael W. Rencheck
Director
Nuclear Engineering and Projects

Sworn to and subscribed before me this 30th day of October, 1998, by Michael W. Rencheck.

Lisa Ann McBride

Signature of Notary Public
State of Florida

NOTARY PUBLIC STATE OF FLORIDA
LISA ANN MCBRIDE
Notary Public, State of Florida
My Comm. Exp. Oct. 25, 1999
Comm. No. CC 505458

LISA ANN McBRIDE

(Print, type, or stamp Commissioned Name of Notary Public)

Personally Known X -OR- Produced Identification ______

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72

ATTACHMENT A

LICENSE AMENDMENT REQUEST #245, REVISION 0

Revision to Licensing Basis - Methodology Change for Boraflex Degradation

License Amendment Request,
No Significant Hazards Consideration Evaluation,
and Environmental Impact Evaluation

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72

LICENSE AMENDMENT REQUEST #245, REVISION 0
REVISION TO LICENSING BASIS - METHODOLOGY CHANGE FOR BORAFLEX DEGRADATION

LICENSE DOCUMENTS INVOLVED: Final Safety Analysis Report (FSAR)
Improved Technical Specifications (ITS) Bases

PORTIONS:
FSAR Section 9.3.2.6.1, "Spent Fuel Pools Supplemental Cooling"
FSAR Section 9.6.1.2.2, "Spent Fuel Storage"
FSAR Section 9.6.2.4, "Safety Provisions"
FSAR Table 9-14, "Fuel Storage Racks Subcriticality Margin - 5.0% Enrichment"
ITS Bases B 3.7.14, "Spent Fuel Pool Boron Concentration"
ITS Bases B 3.7.15, "Spent Fuel Assembly Storage"

SUMMARY OF CHANGES:

This submittal requests NRC review and approval for a change in analysis methodology to calculate criticality in the Spent Fuel (SF) Pool B.

Changes to add reference to, and discussion of the new criticality analysis associated with Boraflex degradation to the FSAR and ITS Bases will be used as a means to document the approval of the analysis methodology change. These proposed revisions are provided on an information basis to aid in the review process.

BACKGROUND:

The calculations provided in the current Crystal River Unit 3 (CR-3) licensing basis (Reference 1) cannot accommodate current and projected Boraflex losses and still ensure SF Pool B will remain at less than or equal to 0.95 k-effective when fully loaded and flooded with unborated water. Therefore, a new criticality analysis (Attachment C) is being proposed. The new analysis demonstrates that the burnup/enrichment curves of current CR-3 ITS have sufficient margin to accommodate up to a 20% loss in Boraflex neutron absorption capability. This new analysis uses current NRC accepted calculational methodologies that allow a more realistic set of assumptions than the previous analysis.

The principal calculational differences are:

- The new analysis uses a 238 group cross-section set instead of the earlier 123 group set.
- The new analysis combines uncertainties statistically rather than assuming all worst case tolerances reside simultaneously.
- The new analysis assumes random distribution of gaps rather than axially aligned gaps at the midplane.
- The new analysis includes a 5% uncertainty factor for depletion calculations.
- The new analysis uses newer versions of CASMO and KENO computer codes.

Similar methodology to that being proposed in this LAR has been utilized in part and licensed for SF Pool A at CR-3 (Reference 6), Waterford Unit 3 (Reference 7), and Three Mile Island Unit 1 (Reference 8).

CHANGES TO FSAR AND ITS BASES SECTIONS:

FSAR Section 9.3.2.6.1, "Spent Fuel Pools Supplemental Cooling"
FSAR Section 9.6.1.2.2, "Spent Fuel Storage"
FSAR Section 9.6.2.4, "Safety Provisions"
FSAR Table 9-14, "Fuel Storage Racks Subcriticality Margin - 5.0% Enrichment"
ITS Bases B 3.7.14, "Spent Fuel Pool Boron Concentration"
ITS Bases B 3.7.15, "Spent Fuel Assembly Storage"

Description of Change:

Proposed FSAR and ITS Bases Sections will be changed to read:

FSAR Section 9.3.2.6.1, "Spent Fuel Pools Supplemental Cooling"

Last sentence of the paragraph below added:

Thermal hydraulic analysis of Spent Fuel Pools A & B concluded that natural circulation flow is adequate to preclude local boiling by a large margin. The analysis for Pool A, "Thermal-Hydraulic Design Analysis Report for Crystal River Unit 3 High Density Fuel Storage Racks" was submitted to the NRC by FPC letter dated March 3, 1978. The analysis for Pool B, "CR-3, Spent Fuel Storage Pool B, Rack Modification, Safety Analysis Report," was submitted to the NRC by FPC letter dated October 31, 1989. Reactivity concerns are addressed by a more recent analysis (see section 9.6.1.2.2).

FSAR Section 9.6.1.2.2, "Spent Fuel Storage"

Last sentence of the paragraph below modified:

> Fuel stored in the high density poison racks of Spent Fuel Pool A have a center-to-center distance of 10.5 inches. Fuel stored in the high density racks in Spent Fuel Pool B have a center-to-center distance of 10.60 inches for Region 1, and a 9.17 inches center-to-center distance for Region 2. Both regions utilize a neutron absorbing material (BORAFLEX). In either case, spacing is sufficient to maintain a subcritical condition when wet. Control Rod Assemblies (CRA) requiring removal from the reactor are stored in the spent fuel assemblies. Subcriticality margins for the New Storage Vault and the Spent Fuel Storage Pools are noted in Table 9-14. They are based on the reports "Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 with Fuel of 5.0% Enrichment" submitted to the NRC by FPC letter dated March 9, 1995, and "Reanalysis of the Crystal River Pool B Spent Fuel Storage Racks with Boraflex Degradation" submitted to the NRC by FPC letter 3F1098-15.

FSAR Section 9.6.2.4, "Safety Provisions"

Last sentence of the paragraph below added:

> Spent Fuel Pool B utilizes high-density storage racks with a two-region arrangement. Region 1 has 174 locations and Region 2 has 641 locations. The racks are designed to Seismic Category I criteria. The criticality analysis, seismic analysis, and poison material properties are contained in "CR-3 Spent Fuel Storage Pool B Rack Modification, Safety Evaluation Report," submitted to the NRC by FPC letter dated October 31, 1989. As discussed in Section 9.6.1.2.2, additional submittals were made in 1995 to reflect an increase in fuel enrichment to 5.0%. License Amendment No. 151, dated December 15, 1995 authorized the increase to 5.0% enrichment. To demonstrate an allowance for BORAFLEX loss, License Amendment No. *[License Amendment Number]*, dated *[Date approved]*, approved a criticality analysis assuming limited BORAFLEX loss.

FSAR Table 9-14, "Fuel Storage Racks Subcriticality Margin - 5.0% Enrichment"

Changes to the k-effective and Subcriticality Margin to reflect the following (shown respectively):

Pool B - High Density Racks - Region 1 Unborated 0.9179* and 8.9*

Pool B - High Density Racks - Region 2 Unborated 0.9410* and 6.3*

*Assumes 20% loss of BORAFLEX.

ITS B 3.7.14, "Spent Fuel Pool Boron Concentration"

Reference 2 changed to reflect new analysis:

2. Reanalysis of the Crystal River Pool B Spent Fuel Storage Racks With Boraflex Degradation, S. F. Turner, Holtec International, HI-982056, October 20, 1998.

ITS B 3.7.15, "Spent Fuel Assembly Storage"

Reference 2 changed to reflect new analysis:

2. Reanalysis of the Crystal River Pool B Spent Fuel Storage Racks With Boraflex Degradation, S. F. Turner, Holtec International, HI-982056 dated October 20, 1998.

Reason for Request:

FPC has identified that Boraflex samples removed from SF Pool B have a current weight loss of 1.75% in Region 1, and 4.68% in Region 2. The calculations currently providing the licensing basis (Reference 1) for ITS Design Features 4.3.1.1.b, and the burnup/enrichment curve in ITS LCO 3.7.15, cannot accommodate losses of this magnitude and still ensure SF Pool B will remain less than or equal to 0.95 k-effective when fully loaded and flooded with unborated water. The new criticality analysis (Attachment C) shows that the existing ITS LCO 3.7.15 curves will ensure SF Pool B will remain less than or equal to 0.95 k-effective with as much as 20% degradation of the Boraflex neutron absorber material.

Based on the guidance provided in NRC Generic Letter 91-18, Revision 1, FPC has determined that the Boraflex degradation represents a nonconforming condition, but the SF Pool B remains operable. Deficiency Report (DR) 98-3327 was developed to document this decision, and is available for NRC review at the facility.

FPC has evaluated this change to the CR-3 FSAR in accordance with the provisions of 10 CFR 50.59(a)(2), and has determined that, utilizing CR-3 current, NRC-approved, calculational methodologies and assumptions, the Boraflex degradation would result in a reduction of the margin of safety. The proposed criticality analysis uses assumptions and methodologies which demonstrate the margin of safety is preserved, however, they are based on more realistic input assumptions than the currently approved analysis. Therefore, an unreviewed safety question exists.

Justification for Request:

The new criticality analysis being proposed was performed by Holtec International (Attachment C). The proposed analysis provides three areas of improvement: (1) enhancements in computer modeling, (2) determines the potential effect of Boraflex degradation on criticality safety, and (3) confirms configurations for acceptable storage of fuel with enrichments up to 5, plus or minus 0.05%, U-235. This updated evaluation encompasses both Regions 1 and 2 of SF Pool B at CR-3, and considers the potential effects of up to 20% loss of the Boraflex absorber.

The current analyses are based on the existing ITS Figures 3.7.15-2 and 3.7.15-3 which provide the acceptable burnup-enrichment combinations for safe storage of fresh and spent fuel. The Region 1 storage cells are separated by two Boraflex panels with a flux-trap water gap between the two panels, and is designed for fresh fuel of 5.0% enrichment positioned in a checkerboard pattern with spent fuel of specified enrichment-burnup combinations. Region 2 consists of a uniform array of cells composed of a single Boraflex absorber panel between cells designed for spent fuel of specified enrichment-burnup combinations.

The primary differences between the new criticality analysis and the previous analysis are as follows:

- The new analysis uses the full 238 group cross-section set based on the ENDF/BV cross sections, in contrast to the earlier 123 group set based on ENDF/BII cross-sections.

- The parametric evaluations in the new analysis are performed for each of the manufacturing tolerances, and associated reactivity uncertainties are combined statistically rather than assuming that all tolerances were at their "worst" value simultaneously everywhere throughout the racks.

- The previous evaluation acknowledged that the axial distribution in Boraflex gaps was random, but chose to assume 4-inch gaps on only two panels per cell, with all occurring at the fuel midplane. The new analysis calculates the effect of random distribution of 4-inch gaps in all Boraflex panels, which is consistent with effects observed in industry rack blackness tests. Additionally, CR-3 rack design does not lend itself to gap formation.

- In contrast to the factors reducing the calculated reactivity, inclusion of an estimate of the uncertainty in depletion is calculated. This uncertainty was taken as 5% of the reactivity decrement from Beginning of Life, to burnup of interest.

- CASMO-4 and KENO-5a computer codes are used in the new analysis. The previous analysis used previous releases of the codes, CASMO-3 and KENO-4.

Boraflex is known to degrade under the influence of gamma radiation and chemical reaction with free radicals in the pool water. Over the first few years of use, the Boraflex will shrink, typically creating gaps distributed randomly in the axial direction. As the gamma dose increases, the Boraflex panels will slowly begin to deteriorate, losing the neutron absorbing component (B_4C). Although CR-3 Boraflex panel design does not lend itself to gap formation, the new analysis conservatively assumes the presence of 4-inch gaps in all Boraflex panels. The potential reactivity consequences of concurrent loss of up to 20% of the Boraflex were also evaluated.

Results of the new analysis confirm that, for the existing ITS Figures 3.7.15-2 and 3.7.15-3 limits, there is sufficient margin in both Region 1 and Region 2 of the CR-3 SF Pool B storage racks to accommodate both the potential gaps in the Boraflex and the concurrent loss of up to 20% of the Boraflex absorber material. Accident analyses were also performed, and established that for the most serious fuel misloading accident (Region 2), criticality will not be reached. Also, with 350 ppm soluble boron, which is well below the ITS limit of greater than or equal to 1925 ppm, the maximum k-effective is maintained below the ITS limits.

The following describes the methodology used in the new analysis:

The primary criticality analyses were performed with the three-dimensional NITAWL-KENO-5a Monte Carlo code package (Reference 3). NITAWL was used with the 238 group SCALE-4.3 cross-section library and the Nordheim integral treatment for U-238 resonance shielding effects. Benchmark calculations, presented in the analysis, indicate a bias of 0.0030Δk $\pm$0.0012 (95%/95%). Verification calculations for the principal cases were made with the MCNP code (Reference 4), with a bias of 0.0009Δk $\pm$0.0011, as shown in Appendix A of Attachment C. CASMO-4, a two-dimensional deterministic code (Reference 5), was used to evaluate the small reactivity effects of manufacturing tolerances using transmission probabilities. Validity of the CASMO-4 code was established by comparison with KENO-5a and MCNP calculations.

In the geometric model used in the calculations, each fuel rod and each fuel assembly were explicitly described. The calculational model used a 4x4 array of cells with 4-inch gaps in the Boraflex randomly distributed axially. Reflecting boundary conditions effectively defined an infinite radial array of storage cells. In the axial direction, a 30-cm water reflector was used to conservatively describe axial neutron leakage. Each stainless steel box and all associated Boraflex panels were also explicitly described in the calculational model. The cladding material was conservatively assumed to be zirconium. The actual cladding material is Zircaloy, with a greater absorption cross-section that would slightly reduce reactivity.

Monte Carlo (KENO-5a) calculations inherently include a statistical uncertainty due to the random nature of neutron tracking. To minimize the statistical uncertainty of the KENO-5a calculated reactivities, a minimum of 1 million neutron histories was accumulated in each calculation, generally resulting in a statistical uncertainty of about $\pm$ 0.0006Δk (1σ).

Two independent methods of analyses (KENO-5A and MCNP) were used to verify the principal calculations. In addition, these calculations serve to validate the CASMO-4 code, since CASMO-4 is a two-dimensional code and cannot be directly validated against critical experiments. Results of these code comparison calculations are described in the analysis. These results are considered to be in good agreement, confirming the basic KENO-5a, MCNP and CASMO-4 calculations.

To assure the true reactivity is bounded by calculated reactivity, the following conservative criteria or assumptions were used in the analysis:

- The racks contain the most reactive fuel authorized to be stored, without any control rods or burnable poison.

- The moderator is pure, unborated water at a temperature within the design basis range corresponding to the highest reactivity.

- Criticality safety analyses are based upon an infinite radial array of cells (i.e., no credit is taken for radial neutron leakage).

- Neutron absorption in minor structural members is neglected (e.g., spacer grids are replaced by water).

- The analyses were based on the enrichment-burnup combinations in the current ITS and current rack design details.

- The analyses assumed the most reactive combinations of enrichment and burnup allowed for licensed fuel designs at CR-3.

Codes, standards, and regulations utilized in the new analysis include:

- General Design Criterion 62, 10 CFR 50 Appendix A, Prevention of Criticality in Fuel Storage and Handling.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage.

- USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.

- USNRC Regulatory Guide 1.13, Spent Fuel Storage Facility Design Basis, Revision 2 (proposed), December 1981.

- ANSI-8.17-1984, Criticality Safety Criteria for the Handling, Storage and Transportation of LWR Fuel Outside Reactors.

- L. Kopp, "Guidance On The Regulatory Requirements For Criticality Analysis Of Fuel Storage At Light-Water Reactor Power Plants", USNRC Internal Memorandum L. Kopp to Timothy Collins, August 19, 1998.

NO SIGNIFICANT HAZARDS CONSIDERATION EVALUATION

An evaluation of this proposed LAR has been performed in accordance with 10 CFR 50.91(a)(1) regarding significant hazard considerations, using the standards in 10 CFR 50.92(c). A discussion of these standards as they relate to this LAR follows:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

No. The two possible accidents are: (1) criticality during normal storage and (2) criticality due to a misloaded fuel assembly during handling fuel. Each are discussed below:

(1) Criticality during normal storage.

For criticality during normal storage to occur, there must be a loss of negative reactivity since an addition of positive reactivity is not possible without fuel movement. A loss in negative reactivity could result only from reduction in Boraflex inventory below that needed to meet the design basis. The proposed criticality analysis for Spent Fuel Pool B demonstrates that Spent Fuel Pool B is capable of maintaining the design basis requirement of k-effective less than or equal to 0.95 when flooded with unborated water and with a loss of up to 20% of the Boraflex absorber material. Therefore, allowing up to 20% Boraflex loss with the new analysis does not significantly increase the probability of an accident previously evaluated.

(2) Criticality during fuel handling.

Criticality during fuel handling could occur due to loss of negative reactivity, or the addition of positive reactivity. Loss of negative reactivity could result from loss of Boraflex as discussed above.

Addition of positive reactivity would result from the misloading of fuel in a fashion not in accordance with ITS LCO 3.7.15, such as the misloading of a fresh 5.05% enriched fuel assembly into Region 2 or side-by-side with another fresh fuel assembly in Region 1. The minimum required boron concentration of ITS LCO 3.7.14 and CR-3 FSAR 9.3.2.1.2 are intended to compensate for just such an accident. Consistent with the double-contingency principle, a boron dilution is not required to be considered concurrent with a misloaded new fuel assembly (bases of ITS LCO 3.7.14). The use of a new calculational method will not increase the probability of fuel assembly misloading. A boron dilution event without an accompanying misloaded fuel assembly is not impacted by the new criticality analysis, since the design basis allows for unborated water for normal storage conditions.

Therefore, since the proposed criticality analysis does not increase the probability of a misloaded fuel assembly, the probability of an occurrence of an accident previously evaluated is not significantly increased.

Boraflex is credited with preventing inadvertent criticality. It is not credited with mitigating the effects, or dose consequences, to the public or to plant personnel from an inadvertent criticality. The criticality analysis does not affect or mitigate the dose consequences to the public or plant personnel from an inadvertent criticality.

There are no other SAR accidents that could be affected.

Therefore, the use of the proposed criticality analysis, does not significantly increase the consequences of an accident previously evaluated.

(2) *Create the possibility of a new or different kind of accident from any accident previously evaluated.*

No. The only purpose, or function, of Boraflex is reactivity control. Therefore, the use of the proposed criticality analysis can only result in reactivity related accidents, such as an inadvertent criticality. Though a spent fuel pool criticality accident is not discussed in detail, a calculation to ensure such an accident could not occur is referenced by both FSAR 9.3 and 9.6. Therefore, this is an accident already discussed by the SAR and dependence on a new criticality analysis does not create the possibility of an accident of a new or different kind than any previously evaluated.

(3) *Involve a significant reduction in a margin of safety.*

No. The proposed analysis demonstrates that the safety function and design basis are met even for a Boraflex loss of up to 20%. Though the proposed criticality analysis methodology is more realistic, and has been licensed at other sites, it is less conservative than the existing, NRC approved analysis that is currently part of the CR-3 licensing basis. Additionally, it permits operation with a greater loss of Boraflex than the existing analysis.

The current licensing basis, BAW-2209, "Crystal River Unit 3 Spent Fuel Storage Pool Criticality Analysis", provides the analytical basis of both ITS LCO 3.7.14 and LCO 3.7.15. This analysis uses very conservative assumptions and methodologies, and results in very little margin remaining for identified Boraflex loss. The margin of safety, although less than previously evaluated, is not significantly reduced with reliance on the current criticality analysis. The margin of safety is restored with use of the proposed criticality analysis. Therefore, the margin of safety is not significantly reduced with use of the proposed criticality analysis.

ENVIRONMENTAL IMPACT EVALUATION

10 CFR 51.22(c)(9) provides criteria for and identification of licensing and regulatory actions eligible for categorical exclusion from performing an environmental assessment. A proposed amendment to an operating license for a facility requires no environmental assessment if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration, (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released off-site, or (3) result in a significant increase in individual or cumulative occupational radiation exposure. FPC has reviewed this license amendment and has determined that it meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(c), no environmental impact statement or environmental assessment need to be prepared in connection with the issuance of the proposed license amendment. The basis for this determination is as follows:

1. The proposed license amendment does not involve a significant hazards consideration as described previously in the evaluation.

2. As discussed in the significant hazards evaluation, this change does not result in a significant change or significant increase in the radiological doses for any Design Basis Accident. The proposed license amendment does not result in a significant change in the types or a significant increase in the amounts of any effluents that may be released off-site. FPC has concluded that there will not be a significant increase in the types or amounts of any effluents that may be released off-site and does not involve irreversible environmental consequences beyond those already associated with The Final Environmental Statement.

3. The proposed license amendment does not result in a significant increase to the individual or cumulative occupational radiation exposure because this change of methodology within the criticality analysis could not increase occupational radiation exposure.

REFERENCES:

1. W.A. Witkopf and L.A. Hassler, "Crystal River Unit 3 Spent Fuel Storage Pool Criticality Analysis," BAW-2209, Revision 1, February 1, 1995

2. Holtec Report, HI-982056, "Reanalysis of The Crystal River Pool B Spent Fuel Storage Racks with Boraflex Degradation," October 20, 1998

3. R. M. Westfall, et. Al., "NITAWL-S: Scale System Module for Performing Resonance Shielding and Working Library Production" in SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation, NUREG/CR-0200, 1979

   L.M. Petrie and N.F. Landers, "KENO Va. An Improved Monte Carlo Criticality Program with Subgrouping" in SCALE: A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation, NUREG/V-0200, 1979

4. J.F Briemeister, Ed., "MCNP – A General Monte Carlo N-Particle Transport Code, Version 4A," Los Alamos National Laboratory, LA-12625-M (1993)

5. A. Ahlin, M. Edenius, H. Haggblom, "CASMO-A Fuel Assembly Burnup Program," AE-RF-76-4158, Studsvik report (propriety)

   A. Ahlin and M. Edenius, "CASMO- A Fast Transport Theory Depletion Code for LWR Analysis," ANS Transactions, Vol. 26, p. 604, 1977

   M. Edenius et al., "CASMO4 Benchmark Against Critical Experiments," Studsvik Report SOA-94/13 (Proprietary)

   M. Edenius et al., "CASMO4, A Fuel Burnup Program, Users Manual" Studsvik Report SOA/95/1

6. NRC to FPC letter, 3N1295-08, dated December 15, 1995, "Issuance of Amendment Re: Fuel Enrichment Increase (TAC No. M91536)"

7. NRC to Entergy Operations, Inc. letter, dated July 10, 1998, "Issuance of Amendment No. 144 to Facility Operating License. NPF-38 – Waterford Steam Electric Station, Unit 3 (TAC No. M98325)"

8. NRC to GPU Nuclear Corporation letter, dated February 17, 1993, "Issuance of Amendment (TAC No. M84596)," Amendment No. 170 to Facility Operating License. DPR-50 – Three Mile Island Nuclear Station, Unit 1

FLORIDA POWER CORPORATION
CRYSTAL RIVER UNIT 3
DOCKET NO. 50-302/LICENSE NO. DPR-72

ATTACHMENT B

LICENSE AMENDMENT REQUEST #245, REVISION 0
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Proposed Revisions to Final Safety Analysis Report and Improved Technical Specification Bases in Strikeout/Highlighted Format

b. Core fuel assemblies will not be transferred to the Spent Fuel Pool while this alignment is in effect.

c. Fuel movement within the Spent Fuel Pool is suspended.

d. Main CI loop is sampled for signs of radioactivity at least once every 24 hours (RM-L3 is secured when CI supplies cooling water and SW pumps are secured).

The arrangements to be made to restore operation of heat removal equipment for the spent fuel pool in the unlikely event all cooling were lost would depend upon the nature of the failure. When a full-core is discharged under the conditions of Section 9.3.1, the SF System will maintain the spent fuel storage pool temperature below 160°F. Failure of a single component in this system will not permit uncovering of the stored spent fuel under normal operating conditions since the system is designed with redundant components. Analyses have shown that considerable time is required to raise the temperature of the pool water to temperatures greater than 190°F during which cooling can be restored.

For the case of a failure of one SF System train with a full-core offload in the pool, the pool temperature could exceed 210°F, if supplemental cooling were not available. Starting with the pool temperature at 160°F, the minimum time for the pools to reach 190°F is approximately 8 hours.

Eight hours is sufficient time to provide an alternate power source, using temporary wiring in the event of failure of the electrical power supply to both spent fuel pump motors.

In the event the coolant flow through the heat exchanger is lost through some massive failure of the SW System to the Spent Fuel Cooling System, eight hours is also sufficient time to repair the damage or provide an alternate source of water to the affected heat exchangers, such as the Plant Fire Protection System, using temporary connections and hoses or by utilizing the Decay Heat System (DH) or CI System which are permanently piped and valved. This position was accepted by the NRC in the safety evaluation that accompanied License Amendment No. 134, dated April 16, 1991.

The effects of using the BWST for supplemental cooling have been analyzed. During Mode 5 or 6 of Refueling, the BWST may be valved into either SFP-1A or SFP-1B pump suction to supply water to the pools. SFP-2 may be utilized during any operational mode. This arrangement provides a much larger heat sink resulting in extended times for the fuel pools to attain a temperature of 190°F. The inclusion of the BWST water decreases the pool temperature for the first several hours of the transient.

Thermal hydraulic analysis of Spent Fuel Pools A & B concluded that natural circulation flow is adequate to preclude local boiling by a large margin. The analysis for Pool A, "Thermal-Hydraulic Design Analysis Report for Crystal River Unit 3 High Density Fuel Storage Racks" was submitted to the NRC by FPC letter dated March 3, 1978. The analysis for Pool B, "CR-3, Spent Fuel Storage Pool B, Rack Modification, Safety Analysis Report," was submitted to the NRC by FPC letter dated October 31, 1989. Reactivity concerns are addressed by a more recent analysis (see section 9.6.1.2.2).

As stated earlier, the spent fuel pool is designed to withstand tornado generated missiles. Because of the lack of substantiated evidence of the occurrence of any type of pool suction due to a three psi pressure drop caused by a tornado, and with substantial analytical evidence (Ref. 1) that it would not occur, FPC has assumed that the amount of water removal, if any, will not prevent the water contained from maintaining its normal protection capability. Therefore, dewatering of the fuel pool due to high wind action is not a cause of concern.

b. With fuel of the highest anticipated reactivity in place and assuming the optimum hypothetical low density moderation (i.e., fog or foam), the maximum reactivity shall not exceed a k_{eff} of 0.98.

The evaluation confirms that the New Fuel Storage Pit can safely accommodate 54 fuel elements of 5.0 weight % enrichment provided certain storage locations remain empty of fuel. The New Fuel Storage Pit has two rows (rows 4 and 8) physically blocked to ensure reactivity limits are not exceeded. Subcriticality margins for the New Fuel Storage Pit are in Table 9-14 and are based on the report "Criticality Safety Evaluation of the Crystal River Unit 3 New Fuel Storage Vault With Fuel of 5% Enrichment" submitted to the NRC by FPC letter dated March 9, 1995.

9.6.1.2.2 Spent Fuel Storage

Both of the Spent Fuel Pools are constructed of reinforced concrete and lined with stainless plate, and are located in the fuel handling area of the Auxiliary Building. Verification of the integrity of Spent Fuel Pools A & B under all possible conditions in light of the increased loads due to the high density rack design is included in GAI Report No. 1949, "Investigation on the Structural Safety of the Spent Fuel Pool Due to Installation of High Capacity Fuel Racks." This report was submitted to the NRC by FPC letter dated January 9, 1978. The Spent Fuel Storage Racks and the New Fuel Storage Pit are designed to Seismic Class I requirements.

Fuel stored in the high density poison racks of Spent Fuel Pool A have a center-to-center distance of 10.5 inches. Fuel stored in the high density racks in Spent Fuel Pool B have a center-to-center distance of 10.60 inches for Region I and a 9.17 inches center-to-center distance for Region II. Both regions utilize a neutron absorbing material (BORAFLEX). In either case, spacing is sufficient to maintain a subcritical condition when wet. Control Rod Assemblies (CRA) requiring removal from the reactor are stored in the spent fuel assemblies. Subcriticality margins for the New Storage Vault and the Spent Fuel Storage Pools are noted in Table 9-14. They are based on the reports "Criticality Safety Evaluation of the Pool A Spent Fuel Storage Racks in Crystal River Unit 3 with Fuel of 5.0% Enrichment" ~~and "Crystal River Unit 3 Spent Fuel Storage Pool B Criticality Analysis"~~ submitted to the NRC by FPC letter dated March 9, 1995**, and "Reanalysis of the Crystal River Pool B Spent Fuel Storage Racks with Boraflex Degradation" submitted to the NRC by FPC letter 3F1098-15.**

9.6.1.3 Fuel Transfer Tubes

The Fuel Transfer Tubes (FHX-1A, FHX-1B) are horizontal tubes are provided to convey fuel and other materials between the Reactor Building and the fuel handling area of the Auxiliary Building. These tubes contain tracks for the fuel transfer carriage, gate valves in the fuel handling area of the Auxiliary Building, and a flanged closure on the Reactor Building. The Fuel Transfer Tubes connect the Spent Fuel Pool to the Fuel Transfer Canal at the lower depth, where space is provided for the rotation of the fuel transfer carriage basket containing a fuel assembly. That portion of the Fuel Transfer Tubes and penetrations which are designed and utilized as the Reactor Building boundary is designed to the Seismic Class I criteria.

9.6.1.4 Fuel Transfer Canal

The Fuel Transfer Canal is a passageway in the Reactor Building extending from the reactor vessel boundary to the Reactor Building wall. It is formed by an upward extension of the primary shield walls. The enclosure is a reinforced concrete structure lined with stainless steel plate to form a canal above the reactor vessel which is filled with borated water for refueling.

The deep portion of the Fuel Transfer Canal is used for storage of the reactor vessel internals, Core Support Assembly, and Plenum Assembly during refueling, however, the deep portion is not large enough to support all these components at any one time.

Fuel Storage Racks," submitted to the NRC by FPC letter dated January 18, 1979, and structural report entitled "Structural Analysis Design Report for Crystal River Unit 3 High Density Storage Racks," submitted to the NRC by FPC letter dated March 3, 1978. The physical properties of the poison material, B_4C, and the associated verification testing performed has been provided to the NRC by FPC letter dated June 29, 1979.

Spent Fuel Pool B utilizes high density storage racks with a two region arrangement. Region 1 has 174 locations and Region 2 has 641 locations. The racks are designed to Seismic Category I criteria. The criticality analysis, seismic analysis, and poison material properties are contained in "CR-3 Spent Fuel Storage Pool B Rack Modification, Safety Evaluation Report," submitted to the NRC by FPC letter dated October 31, 1989. As discussed in Section 9.6.1.2.2, additional submittals were made in 1995 to reflect an increase in fuel enrichment to 5.0%. License Amendment No. 151, dated December 15, 1995 authorized the increase to 5.0% enrichment. **To demonstrate an allowance for BORAFLEX loss, License Amendment No. [Ammendment Number], dated [Date Approved], approved a criticality analysis assuming limited BORAFLEX loss.**

The fuel storage racks are designed so that it is impossible to insert fuel assemblies in other than the prescribed locations, thereby insuring the necessary spacing between assemblies. Fuel handling and transfer containers are also designed to maintain an eversafe geometric array. Under these conditions, a criticality accident during refueling or storage is not considered credible.

Fuel handling equipment is designed to minimize the possibility of mechanical damage to the fuel assemblies during transfer operations. If fuel damage should occur, the amount of radioactivity reaching the environment will be significantly below regulatory limits. Radiological consequences of a fuel handling accident are analyzed in Chapter 14. Structural analysis of accidents involving the high density racks is provided by the report entitled "Structural Analysis Design Report for Crystal River Unit 3 High Density Storage Racks," submitted to the NRC by FPC letter dated March 3, 1978.

All spent fuel assembly transfer operations are conducted underwater. The water level in the Fuel Transfer Canal provides a minimum of 8 feet of water over the active fuel line of the spent fuel assemblies during movement from the core into storage to limit radiation at the surface of the water to less than 2 mrem/hour. The fuel storage racks are designed to have a minimum of 23 feet of water shielding over stored assembly's fuel rods.[TSA149] The minimum depth of the water over the fuel assemblies and the thickness of the concrete walls of the Spent Fuel Pools are sufficient to limit the maximum continuous radiation levels in the working area to 2 mrem/hour or less.

Radiation monitors are provided in fuel handling areas as follows:

RM-A2	=	Auxiliary Building main exhaust
RM-A4	=	Fuel handling and spent fuel exhaust duct
RM-A15	=	Spent fuel area
RM-G14	=	Fuel storage area
RM-G15	=	Auxiliary Building Fuel Handling Bridge
RM-G16	=	Reactor Building Fuel Handling Bridge

Water in the reactor vessel is cooled during shutdown and refueling by the Decay Heat Removal System (DH) as described in Section 9.4. Adequate redundant electrical power supply assures continuity of heat removal. The Spent Fuel Pool water is cooled by the Spent Fuel Cooling System described in Section 9.3. A power failure during

TSA149 *Information in this paragraph was added when Technical Specification Amendment 149 was issued. Obtain Manager Nuclear Licensing concurrence before changing.*

TABLE 9-14 Fuel Storage Racks Subcriticality Margin - 5.0% Enrichment

Condition	K_{eff}	Subcriticality Margin (% Δk/k)
New Fuel Storage Pit		
Flooded	0.948	5.5
Optimum (worst case) Moderation	0.978	2.2
Spent Fuel Racks		
Pool A - High Density Racks Unborated	0.9435	6.0
Pool B - High Density Racks - Region 1 Unborated	~~0.9288~~ 0.9179*	~~7.7~~ 8.9 *
Pool B - High Density Racks - Region 2 Unborated	~~0.9499~~ 0.9410*	~~5.3~~ 6.3 *

* Assumes 20% loss of BORAFLEX.