ATTACHMENT 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

CONTROL ELEMENT ASSEMBLIES

5.3.2 The reactor core shall contain 73 full length and no part length control element assemblies. The control element assemblies shall be designed and maintained in accordance with the original design provisions contained in Section 4.2.3.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

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 5.2 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements,

b. For a pressure of 2485 psig, and

c. For a temperature of 650°F, except for the pressurizer which is 700°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 11,100 ± 180 cubic feet at a nominal T_{avg} of 567°F.

5.5 EMERGENCY CORE COOLING SYSTEMS

5.5.1 The emergency core cooling systems are designed and shall be maintained in accordance with the original design provisions contained in Section 6.3 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.6 FUEL STORAGE

CRITICALITY Replace Section 5.6.1 with Attachment A

~~5.6.1 The new fuel storage racks are designed and shall be maintained with a center-to-center distance of not less than 21 inches between fuel assemblies placed in the storage racks. The spent fuel storage racks are designed and shall be maintained with a center-to-center distance of not~~

CRITICALITY (Continued)

less than 12.53 inches between fuel assemblies placed in the storage racks. These spacings ensure a K_{eff} equivalent to ≤ 0.95 with the storage pool filled with unborated water. The K_{eff} of ≤ 0.95 includes the conservative assumptions as described in Section 9.1 of the FSAR. In addition, fuel in the storage pool shall have a U-235 loading of $\leq$ 41.45 grams of U-235 per axial centimeter of fuel assembly ($\leq$ an enrichment of 3.7 weight percent U-235).

DRAINAGE

5.6.2 The fuel pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 56 feet.

CAPACITY

5.6.3 The spent fuel pool is designed and shall be maintained with a storage capacity limited to no more than 728 fuel assemblies.

5.7 SEISMIC CLASSIFICATION

5.7.1 Those structures, systems and components identified as seismic Class I in Section 3.2.1 of the FSAR shall be designed and maintained to the original design provisions contained in Section 3.7 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.8 METEOROLOGICAL TOWER LOCATION

5.8.1 The meteorological tower location shall be as shown on Figure 5.1-1.

5.9 COMPONENT CYCLE OR TRANSIENT LIMITS

5.9.1 The components identified in Table 5.9-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.9-1.

ATTACHMENT A

5.6 FUEL STORAGE

CRITICALITY

5.6.1 a. The spent fuel storage racks are designed and shall be maintained with:

1. A k_{eff} equivalent to less than or equal to 0.95 with the storage pool filled with unborated water, which includes the conservative assumptions as described in Section 9.1 of the FSAR.

2. A center-to-center distance of not less than 12.53 inches between fuel assemblies placed in the storage racks.

3. A boron concentration greater than or equal to 1720 ppm. In addition, fuel in the storage pool shall be a U-235 enrichment of less than or equal to 4.0 weight percent.

b. The new fuel storage racks are designed for dry storage of unirradiated fuel assemblies having a U-235 enrichment less than or equal to 4.0 weight percent, while maintaining a k_{eff} of less than or equal to 0.98 under the most reactive condition.

Safety Evaluation for a Proposed Change to the St. Lucie Unit 1 Technical Specification 5.6.1 for Increasing the Maximum Fuel Enrichment to 4.0 w/o U-235.

BACKGROUND

In 1977 a request to amend the St. Lucie Unit 1 Operating License for increased spent fuel storage was submitted to the NRC. By letter dated March 28, 1978, the Commission approved Amendment 22 to the Facility Operating License DPR-67 which allowed the modification to the spent fuel pool storage facility.

The modification consisted of reracking the spent fuel pool with (HI-CAP™) fuel storage racks designed and manufactured by Combustion Engineering. These new racks increased the storage capacity from 310 fuel assemblies to 728 fuel assemblies in the spent fuel pool.

The safety evaluation performed in support of the request to amend the St. Lucie Unit 1 Operating License to allow reracking of the spent fuel pool addressed the following:

1. Structural and Seismic Analysis

2. Nuclear Criticality Analysis

3. Thermal Hydraulic Analysis

4. Accident Analyses

5. Radiation Exposures

The criticality analysis was performed for a 3.7 w/o U-235 fuel enrichment (linear loading of 41.45 gms/cm U-235).

It was determined that the proposed modification to the Unit 1 spent fuel pool would be acceptable because the results of the above analysis were within acceptable limits. As a result, the Criticality Technical Specification (5.6.1) was updated to reflect the center to center spacing of the modified racks (12.53 inches minimum spacing) and the maximum allowable enrichment of 41.45 grams of U-235 per axial centimeter of fuel assembly.

A request was forwarded to the NRC in October 1979 to amend the Unit 1 operating license for increasing the fuel assembly enrichment Technical Specification (5.3.1) from 3.1 w/o U-235 to a maximum enrichment of 3.7 w/o U-235. The analysis submitted in support of this License amendment consisted of performing criticality analyses of the high capacity spent fuel racks, new fuel racks, fuel inspection elevator, upender and fuel transfer tube. The results of that safety analysis showed that with a limiting feed enrichment of 3.7 w/o U-235 the multiplication factor for the various structures analyzed did not exceed the limiting multiplication factors of 0.98 for optimum moderation and 0.95 for fully flooded conditions for the new fuel storage racks and 0.95 for the spent fuel racks and fuel handling structures. It was decided, based on discussions between NRC and FPL staff, that the specification of reload fuel enrichment alone does not uniquely determine nor limits, the values of reactor core parameters important to safety. Therefore, the decision was made to delete the enrichment limit of fuel to be used in the reload core (which used to be 3.1 w/o U-235) from

Technical Specification 5.3.1. The 3.7 w/o U-235 enrichment limit was added to the fuel storage Tech. Spec. (5.6.1). This addition to Tech. Spec. 5.6.1 did not change the existing limit but rather clarified it in terms of fuel enrichment (weight percent). By letter dated January 23, 1980, the Commission approved Amendment 34 to the Facility Operating License DPR-67 which deleted the 3.7 w/o U-235 referenced in Tech. Spec. 5.3.1 and added the 3.7 weight percent value to the Criticality Technical Specification (5.6.1).

With this application, FPL is requesting approval to increase the maximum enrichment specification of the Criticality Technical Specification (5.6.1) at St. Lucie Unit 1 to 4.0 w/o U-235 (axial U-235 loading of 43.91 g/cm). The motivation for this proposed increase is to allow increased flexibility in fuel management and to accommodate storage of higher enrichments for possible use in future cycles. The analysis performed in support of this proposed change can be found in Appendix 1. A summary of the results of that analysis are discussed in the next section of this report.

Safety Evaluation

The analysis of the proposed increase in fuel enrichment has been accomplished using current accepted codes and standards as specified in the Safety Analysis Report. Calculations performed for the handling and storage of 4.0 w/o U-235 enriched fuel assemblies indicate that the applicable criticality acceptance criteria are met. The evaluation was performed for natural uranium axial blanket fuel with a maximum central fuel region enrichment of 4.0 w/o U-235. It is important to note that the natural uranium axial blankets were neglected and hence, the analysis is bounding for both 4.0 w/o U-235 enriched fuel with and without axial blankets.

Calculations performed for the spent fuel racks indicate that under worst credible conditions, the neutron multiplication factor is 0.918 keff at the 95% confidence level. This value is considerably lower than the 0.95 safety criteria limit as specified in Reference 2 of the Safety Analysis Report. Calculations performed for the new fuel storage area at various degrees of moderation (including full flooding) indicate that the limiting keff occurs for a moderator void fraction between 0.90 and 0.95 and is estimated to be about 0.925 at the 95% confidence level. This value is also considerably lower than the safety criteria limit of 0.98 specified in ANS-N18.2.

Criticality calculations were also performed for the fuel handling structures. The most reactive situation, i.e. the one that produced the highest reactivity, involved the fuel elevator when it was assumed that one assembly was in the elevator and one additional assembly was located four (4) inches edge to edge from the elevator assembly. The resulting keff from this scenario is 0.924 at the 95% confidence level. Based on this, it is concluded that the fuel elevator, upender and transfer tube will meet the safety criteria limit of keff $\leq$ 0.95. The No Significant Hazards Evaluation of this proposed Technical Specification change can be found in the next section of this evaluation.

References

1. R. E. Uhrig (FPL) to V. Stello (NRC) Re: St. Lucie Unit 1, Docket No. 50-335, Proposed Amendment to Facility Operating License DPR-67, L-77-273, dated 8/31/77.

2. R. E. Uhrig (FPL) to V. Stello (NRC) Re: St. Lucie Unit 1, Docket No. 50-335, Proposed Amendment to Facility Operating License DPR-67, L-79-282 dated 10/4/79.

NO SIGNIFICANT HAZARDS

EVALUATION

No Significant Hazards Evaluation

With this application, FPL is requesting approval to increase the maximum U-235 enrichment and linear loading specified in Technical Specification (5.6.1) from the currently licensed ≤ 3.7 w/o U-235 (axial loading of ≤ 41.45 g/cm) to ≤ 4.0 w/o U-235 (axial loading ≤ 43.91 g/cm) at St. Lucie Unit 1. An evaluation of this request has been performed to demonstrate that no significant hazards consideration exists, based on a comparison with the criteria of 10CFR50.92(C).

The following evaluation demonstrates (by reference to the analysis contained in the attached Safety Analysis Report) that the proposed amendment to increase the enrichment specification does not exceed any of the three significant hazards consideration.

1. The requested change does not increase the probability or consequences of accidents previously analyzed. Since the configuration of the plant and the mode of operation remain unchanged, the probability of accidents previously analyzed remains unchanged.

   FPL has identified the following potential accident scenarios whose consequences would be affected by the proposed change.

   A. A fuel assembly drop in the spent fuel pool.

   B. Loss of spent fuel pool cooling system and makeup.

   C. Spent fuel cask drop.

For A, the criticality acceptance criterion is not violated as identified in Section 3.3 of the Safety Analysis Report. The radiological consequences of this type of accident are bounded by the fuel handling accident analyzed in the FSAR because this application is not intended for extended burnup operation. In particular, the assumptions used in the FSAR fuel handling accident (i.e. burnup, fractional release, etc) are still bounding for the higher enriched fuel assemblies. Based on this discussion, it is concluded that the proposed amendment will not result in an increase of the probability or consequences from the previously evaluated fuel handling accident.

The consequences of B, "loss of spent fuel cooling system and makeup" will not be affected since this application is not intended to qualify the fuel for extended burnup operation. The increase in U-235 enrichment linear loading will not affect the decay heat characteristics of the fuel assembly or the previous FSAR evaluation (Section 9.1.3) of the loss of spent fuel cooling system and makeup. Based on this, it is concluded that the proposed increase in the U-235 enrichment linear loading will not involve a significant increase in the probability or consequences of an accident previously evaluated.

The consequences of C, "a spent fuel cask drop", will not be affected by an increase in linear loading since this application is not intended to qualify the fuel for extended burnup nor is the configuration of the storage racks being altered. Therefore, the consequences of a cask drop accident are still bounded by the previously evaluated FSAR Chapter 15 cask drop analysis. In conclusion, the proposed amendment will not result in an increase of the probability or consequences of an accident previously evaluated for a cask drop.

Based on the above findings, the proposed amendment to increase the maximum allowable U-235 linear loading and corresponding enrichment does not result in an increase in the probability or consequences of an accident previously evaluated.

2. The requested change does not create the possibility of a new or different kind of accident from any accident previously evaluated because the plant configuration and the manner in which it is operated remain the same. The proposed change does not constitute any change in the procedures for plant operation or hardware. In addition, FPL has evaluated the proposed technical specification changes in accordance with the guidance of the NRC position paper entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", and appropriate Industry Codes and Standards as listed in the Reference section of the Safety Analysis Report. Based on this evaluation, FPL finds that the proposed technical specification change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in a margin of safety. As described in the attached Safety Analysis Report, the new fuel storage rack calculated keff of 0.925 (95% confidence level) is considerably lower than the established acceptance criteria of $\leq$ 0.98 keff. The 0.918 keff (95% confidence

level) calculated for the spent fuel pool and 0.924 keff (95% confidence level) calculated for the fuel handling structures is also considerably lower than the established acceptance criteria of $\leq$ 0.95 keff. It is important to note that the above calculated neutron multiplication factors include all the necessary biases and uncertainties.

As noted above, the required acceptance criteria ($\leq$ 0.98 keff under optimum moderation conditions and $\leq$ 0.95 under fully flooded conditions for the new fuel storage racks, $\leq$ 0.95 keff for the spent fuel pool and fuel handling structures) have been adhered to in the criticality analysis performed in support of this proposed technical specification change. Specifically the 0.02 Δ keff and 0.05 Δ keff criticality margin of safety required for the new fuel storage area under optimum moderation and fully floorded conditions respectively, and the 0.05 Δ keff criticality margin of safety required for the spent fuel storage area and fuel handling structures have been maintained as specified in the attached Safety Analysis Report.

Based on the previous discussion, the proposed amendment to increase the fuel storage U-235 linear loading and corresponding enrichment will not involve a significant reduction in the margin of safety for nuclear criticality.

In summary, FPL has determined that the proposed technical specification change does not involve a significant hazard consideration as discussed in 10CFR50.92. Based on the attached Safety Analysis, it is concluded that the health and safety of the public will not be endangered by the proposed change.

APPENDIX 1