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SUBJECT: Forwards responses to 980408 RAI re proposed license amend for TS change on soluble boron credit at St Lucie,Unit 2.

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FPL

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

May 15, 1998

L-98-132
10 CFR 50.90

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

Re: St. Lucie Unit 2
Docket No. 50-389
Proposed License Amendment: SFP Storage Capacity; Soluble Boron Credit
(TAC No. MA0666), Response to Request for Additional Information

Ref: (1) FPL Letter L-97-325, J. A. Stall to NRC (DCD): Proposed License Amendment, SFP Storage Capacity; Soluble Boron Credit; December 31, 1997.

(2) NRC Letter, W. C. Gleaves to T. F. Plunkett (FPL): REQUEST FOR ADDITIONAL INFORMATION-TECHNICAL SPECIFICATION CHANGE REQUEST FOR SOLUBLE BORON CREDIT-ST. LUCIE PLANT, UNIT 2 (TAC NO. MA0666); April 8, 1998.

Reference 1 is an application for license amendment from Florida Power and Light Company (FPL) which will modify Technical Specification 5.6.1 and associated Figure 5.6-1, and Technical Specification 5.6.3 to accommodate an increase in the allowed spent fuel pool storage capacity.

Reference 2 forwarded a request for additional information (RAI), which is identified as information needed by the NRC staff to complete its review of the amendment request. The questions and FPL's responses are contained in the Attachment to this letter.

Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
Nuclear Engineering

1/1

A001

RSK/EJW/RLD

Attachment

cc: Regional Administrator, Region II, USNRC.
Senior Resident Inspector, USNRC, St. Lucie Plant.
Mr. W.A. Passetti, Florida Department of Health and Rehabilitative Services.

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

Rajiv S. Kundalkar being first duly sworn, deposes and says:

That he is Vice President, Nuclear Engineering, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Rajiv S. Kundalkar

Rajiv S. Kundalkar

STATE OF FLORIDA

COUNTY OF St Lucie

Sworn to and subscribed before me

this 15 day of MAY, 1998

by Rajiv S. Kundalkar, who is personally known to me.

George R Madden

Signature of Notary Public-State of Florida

GEORGE R. MADDEN
MY COMMISSION # CC 56202
EXPIRES: June 17, 2000
Bonded Thru Notary Public Underwriters

Name of Notary Public (Print, Type, or Stamp)

1.0 Introduction and Purpose

On December 31, 1997, Florida Power and Light Company (FPL) submitted a proposed license amendment (PLA) to the NRC that would increase the licensed storage capacity of the St. Lucie Unit 2 spent fuel pool (Reference 1). The criticality analysis prepared in support of this submittal credited the negative reactivity associated with the presence of soluble boron in the spent fuel pool and the negative reactivity resulting from decay of fuel assembly actinides during long term post-irradiation storage. The PLA also addresses the likelihood of an inadvertent fuel pool dilution and the impact of the increased spent fuel pool heat load on the pool structure and the existing pool cooling system. During their review of this PLA, the NRC staff has identified several questions for FPL which are documented in Reference 2.

2.0 Response to NRC Questions

Each question is restated and is followed by FPL's prepared response. Input for responses to Questions 1, 2, 4, 8, 9, 10 and 11 was received from Asea Brown Boveri-Combustion Engineering (ABB-CE). Questions 1 and 2 refer to certain Tables and Figures contained in the criticality analysis (CENPD-387, submitted as Enclosure 1 to Reference 1).

2.1 To what fuel assembly configuration does Table 4 and 5 correspond?

FPL Response to Question 1:

The multiplication factors given in Tables 4 and 5 of the Criticality Safety Analysis (CENPD-387) were calculated for the entire fuel pool assuming it was loaded with 1360 fuel assemblies. The fuel assemblies stored in Region I rack modules are arranged as shown in Figure 9. The remaining fuel assemblies are stored in Region II rack modules as shown in Figure 10. The fuel pool and overall rack geometry, including the currently installed cell blocking devices, is shown in Figure 1 of CENPD-387.

2.2 Were the actual loading patterns shown in Figure 9 and Figure 10 mocked up in the analysis?

FPL Response to Question 2:

Yes. The base pool multiplication factor was calculated for the entire pool, in which each subregion (or fuel assembly class) was represented by its equivalent fresh fuel enrichment. These equivalent enrichments are given in Figures 9 and 10.

2.3 What procedures are in place to assure that these loading patterns are not violated?

FPL Response to Question 3:

2.3.1 Fresh Fuel

Following its receipt and inspection, fresh fuel is stored in the new fuel storage racks. At some time subsequent to receipt but prior to the refueling evolution, fresh fuel is typically transferred into storage locations in Region I of the spent fuel racks. As stated in Technical Specification 5.6.1, Region I of the storage racks is designed to accommodate fresh fuel. Placement of new fuel in the spent fuel storage racks is controlled by Operating Procedure (OP) No. 1610020, Receipt and Handling of New Fuel and CEAs and by Administrative Procedure (AP) No. 2-0010250, Guidelines for Use of the Unit 2 High Density Spent Fuel Racks. Section 8, *"Instructions,"* of OP-1610020, Revision 22, clearly states that new fuel may only be stored in Region I of the spent fuel pool. Following approval of FPL's proposed license amendment and prior to placement of fresh fuel assemblies into the proposed spent fuel pool arrangement, these procedures will be modified as necessary to ensure that the Region I loading pattern is not violated.

2.3.2 Irradiated Fuel

Fuel discharged from the core is transferred to the spent fuel pool using the guidance specified in Pre-Operational Procedure (POP) 3200090, Refueling Operation. Appendix E of this procedure contains a detailed list of the required fuel assembly movements which is updated and independently verified prior to use in each refueling campaign. The fuel assembly move list that makes-up this appendix describes the repositioning of each fuel assembly or assembly insert in detail: it specifies the sequential order of each move, the fuel assembly identifier, initial and final locations of the fuel assembly or assembly insert to be moved and the correct azimuthal orientation of the fuel assembly. A current copy of this procedure and the move list are required to be present at each fuel handling machine location during fuel movements.

During the refueling evolution, the parameters of each fuel assembly move are verified against the planned move sequence to ensure total agreement. All parties to the refueling evolution participate in this process. To document assembly moves as they actually occur, each move and the move characteristics are manually entered into a log for comparison to the move list discussed above.

2.3.3 Physical Inventory Verification

As required by 10 CFR 70.51(d), a physical inventory of the spent fuel pool is conducted at least once per year and following the completion of refueling activities. AP- 0010439, Physical Inventory of Nuclear Fuel Storage Areas, provides detailed guidance on how to perform this inventory. Verification of fuel bundle and insert (CEA) serial numbers is required for all fuel pool locations to which special nuclear material has been added or removed. The results of this verification are compared to the approved fuel pool maps found in the Plant Physics Curve Book.

2.3.4 Fuel Currently in Storage

Although the current configuration of fuel assemblies stored in the Unit 2 spent fuel pool meets the reactivity requirements of the proposed license amendment, the currently stored fuel must be repositioned to comply with the requirements of the revised criticality analysis while accommodating the storage of additional fuel assemblies beyond the current limit of 1076. This fuel repositioning will not be performed during refueling operations but will utilize procedural controls equivalent to those in POP-3200090. As described above, an independent verification of fuel assembly locations in the storage racks will be performed following this fuel and insert repositioning. These repositioned permanently discharged fuel assemblies are unlikely to be further handled prior to initiation of fuel cask loading operations.

The design input upon which procedure step sequences are based will be provided by an engineering package. This engineering package will be prepared following approval of the license amendment and will include the conditions necessary to meet the requirements of the criticality, boron dilution, and fuel pool cooling analyses.

2.4 If previously used control element assemblies (CEAs) are used for the inserted CEA cases, how is CEA depletion accounted for?

FPL Response to Question 4:

CEA depletion effects have not been included for the following reasons:

a. St. Lucie Unit 2 operating history is essentially unrodded. Only full strength CEAs are considered in the criticality analysis to control the assembly reactivity in the spent fuel pool; these CEAs must have been part of a five finger regulating or shutdown bank, excluding the lead bank. The lead bank contains reduced strength CEAs which are not suitable to control the reactivity in the spent fuel pool. Typically, all CEA banks are fully withdrawn during steady-state operation, with the exception of a weak insertion programming sequence instituted to mitigate guide tube wear. This unrodded operation minimizes the boron depletion in the CEA fingers.

b. The bottom portion of the CEA finger is composed of Silver-Indium-Cadmium. The five finger full strength control rod design includes a non-depleting 12.5 inch Silver-Indium-Cadmium section at the bottom of each finger, plus two half-inch spacers and a 0.6 inch tip piece. The first depletable B_4C pellet is located about 35 cm above the tip of the control rod. At that elevation, the boron reaction rate is a few percent of what it would be at the tip of the CEA, had the CEA contained B_4C over its entire length. Depletion effects 35 cm above the CEA tip are negligible.

c. The CEA tip is in a low importance region. Should the spent fuel pool approach criticality, it will do so under a top peaked flux distribution, since the fissile content of the burned fuel assemblies is higher at the top of the fuel. The top peaked fission rate in the depleted fuel will also drive the fresh fuel axial flux distribution to a top peaked shape. This will reduce the importance of the CEA tip, and will minimize the impact of any potential CEA depletion.

d. During the lifetime of a CEA, the rod worth is measured at the beginning of each cycle, and no depletion effects are discernible. CEA depletion is not modeled in rod worth physics predictions. The accuracy of these rod worth predictions is monitored during the startup testing program for each fuel cycle. After several cycles of irradiation, there is no indication that the CEA worth has been reduced by depletion effects.

2.5 How is inadvertent CEA removal from stored fuel assemblies precluded?

FPL Response to Question 5:

Manipulation of both fuel and CEAs in the spent fuel pool is performed using the spent fuel handling machine hoist. The grapple tool used to handle CEAs has a different design than the grapple tool used to handle fuel assemblies. As a result, a grapple change must be made following any fuel manipulation in order to initiate a CEA repositioning campaign. Additionally, neither fuel assemblies nor fuel assembly inserts stored in the spent fuel pool may be handled or repositioned except as described in specific procedural guidance.

As noted in the response to question 3 above, a physical inventory of the spent fuel pool is periodically performed as directed by AP-0010439. The locations of fuel assemblies containing CEAs and the CEA identifier are noted during this inventory. Upon determination that the as-found spent fuel pool configuration is, in fact, the correct configuration, this arrangement is documented in the plant physics curve book as the new required fuel pool storage arrangement. Following the completion of fuel handling evolutions and inventory verification the spent fuel handling machine is parked over the fuel transfer canal, configured for long term lay up and deenergized.

2.6 How is the presence of CEAs, if required, in stored fuel assemblies assured?

FPL Response to Question 6:

The presence of CEAs in stored fuel assemblies where required is assured by the requirements of the fuel move list from POP-3200090, Refueling Operation, Appendix E, and by additional guidance that will be added to AP-2-0010250, Guidelines for Use of the Unit 2 High Density Spent Fuel Racks. The permissible pool storage locations for irradiated fuel given in Appendix E are developed using fuel assembly initial enrichment values and a conservative determination of the accumulated fuel assembly burnup. Required fuel pool storage location information is developed as a part of the reload design engineering package and is independently verified. The presence of a CEA in a fuel assembly can be visually confirmed at multiple points during fuel assembly handling.

Fresh fuel is typically received on-site and staged to the spent fuel pool prior to the commencement of refueling operations. Most fresh fuel assemblies initially placed in the fuel pool will not contain CEAs. As a result, these unrodded assemblies initially will be placed in alternating cell locations of rows where storage is permissible.

During refueling operations, fuel assemblies temporarily off-loaded from the core will be placed in a Region I storage location consistent with their burnup and initial enrichment. A number of these off-loaded fuel assemblies will contain full strength CEAs; during full core fuel off-loads future versions of Appendix E to POP-3200090, Refueling Operations, will intersperse the most reactive of these assemblies with the fresh fuel discussed above.

2.7 Should the word "restrictive" in revised TS 5.6.1d be "reactive" ?

FPL Response to Question 7:

Other than the required re-numbering of this item, no changes are proposed to Technical Specification 5.6.1d. As a result, the appropriate word in the fourth line of this specification is "reactive".

2.8 For which specific actinides and fission products is burnup credit given?

FPL Response to Question 8:

The fuel assembly depletion calculations were performed with the two-dimensional multi-group transport code DIT, using an 89-group ENDF/B-VI based cross section library. The DIT cross section library includes 21 actinide and 116 fission product nuclides, which contribute to the reactivity loss with burnup. The depletion chains used to generate nuclide concentrations for the spent fuel pool criticality analysis are the same as those used in nuclear design, which have been benchmarked through core follow analysis. The lists of actinides and fission products are given in the following tables. Samarium and Promethium isotopes of mass number 149 are explicitly modeled with a yield due to direct fission (denoted by the "D" suffix in the table) and a yield due to production from other isotopic decay.

Actinides and Fission Products Included in DIT Depletions

with ENDF/B-VI Cross Sections

Actinides

92-U -235	94-PU-242
92-U -236	95-AM-241
92-U -237	95-AM-242
92-U -238	95-AM-242M
93-NP-237	95-AM-243
93-NP-238	96-CM-242
93-NP-239	96-CM-243
94-PU-238	96-CM-244
94-PU-239	96-CM-245
94-PU-240	96-CM-246
94-PU-241	

Fission Products

35-BR- 81	44-RU-105	53-I -129	59-PR-143	63-EU-151
36-KR- 83	44-RU-106	53-I -131	60-ND-142	63-EU-153
36-KR- 85	45-RH-103	53-I -135	60-ND-143	63-EU-154
38-SR- 89	45-RH-105	54-XE-131	60-ND-144	63-EU-155
38-SR- 90	46-PD-104	54-XE-132	60-ND-145	63-EU-156
39-Y - 89	46-PD-105	54-XE-133	60-ND-146	63-EU-157
39-Y - 91	46-PD-106	54-XE-134	60-ND-147	64-GD-154
40-ZR- 91	46-PD-107	54-XE-135	60-ND-148	64-GD-155
40-ZR- 93	46-PD-108	54-XE-136	60-ND-150	64-GD-156
40-ZR- 95	47-AG-109	55-CS-133	61-PM-147	64-GD-157
40-ZR- 96	47-AG-110M	55-CS-134	61-PM-148	64-GD-158
41-NB- 95	47-AG-111	55-CS-135	61-PM-148M	65-TB-159
42-MO-95	48-CD-110	55-CS-136	61-PM-149D	65-TB-160
42-MO- 96	48-CD-111	55-CS-137	61-PM-149	65-TB-161
42-MO- 97	48-CD-113	56-BA-134	61-PM-151	66-DY-160
42-MO- 98	49-IN-115	56-BA-137	62-SM-147	66-DY-161
42-MO- 99	51-SB-121	56-BA-140	62-SM-148	66-DY-162
42-MO-100	51-SB-123	57-LA-139	62-SM-149D	66-DY-163
43-TC- 99	51-SB-125	57-LA-140	62-SM-149	66-DY-164
44-RU-100	51-SB-127	58-CE-141	62-SM-150	67-HO-165
44-RU-101	52-TE-127M	58-CE-142	62-SM-151	
44-RU-102	52-TE-129M	58-CE-143	62-SM-152	
44-RU-103	52-TE-132	58-CE-144	62-SM-153	
44-RU-104	53-I -127	59-PR-141	62-SM-154	

2.9 What additional uncertainties are assumed when credit for actinide and fission product decay are calculated?

FPL Response to Question 9:

No additional uncertainties were considered for actinide and fission product decay. The decay calculations were performed with the DIT code, using the set of actinide and fission product nuclides described in the response to Question 8. The short lived nuclides such as I-131, Xe-131, Pm-149 and NP-239 were decayed for all pool criticality calculations. For long cooling periods, the decay of Pu-241 into Am-241 is by far the most important transition.

Because St. Lucie Unit 2 is operated with an unrodded core, exposures of discharged assemblies are characterized by nearly symmetric axial burnup distributions that have a slight top peaked bias. As a result, there are minor differences between axially uniform isotopic distributions and the actual assembly nuclide distributions. A slightly top peaked Pu-241 distribution will decay into a slightly top peaked Am-241 distribution. This non-uniformity in isotopic distribution results in a small non-conservative reactivity effect relative to the uniform axial distributions that were assumed in decay calculations.

Conservate measures embedded in the model more than compensate for any postulated axial isotopic distribution effect. For instance, the assumption used in the depletion of a constant 1000 ppm soluble boron and 1200° F fuel temperature results in a hardened spectrum which leads to an overprediction of the fissile content and to an overprediction of the reactivity in the spent fuel pool configuration. Typical reactivity effects after a 15 year cooling period are as follows:

Enrichment (w/o)	Exposure (MWD/T)	Conservatism due to nominal ppm & T_{fuel} % delta-k	Non-conservatism due to axial Pu-241 redistribution at 15 years % delta-k	Net Conservatism % delta-k
3.0	22,400	0.6	0.4	0.2
4.5	40,000	0.6	0.5	0.1

These results should not be extrapolated to higher burnup because the maximum exposure of 40,000 MWD/T considered above conservatively bounds the minimum burnup required in Region II after 15 years of cooling time.

2.10 How do the characteristics of the critical experiments used for benchmarking compare with those of the St. Lucie Unit 2 fuel and spent fuel storage racks?

FPL Response to Question 10:

The fuel used at St. Lucie Unit 2 is the 16 x 16 ABB-CE fuel assembly design for 136.7-inch active height cores. The fuel rods consist of sintered UO_2 pellet columns encapsulated in zircaloy; the assumed UO_2 maximum fuel enrichment is 4.5 wt% U-235 and 5.0 wt% for Gd-UO_2 fuel. The fuel assembly storage rack modules are the Combustion Engineering, Inc. monolithic design and employ 0.135-inch thick SS-304 box walls; in the Region I modules, a 0.188-inch thick SS-304 L-insert is employed in each storage cell.

Selected lattices from two experimental programs were employed in the benchmarking analyses: (1) the Babcock & Wilcox program "Critical Experiments Supporting Close Proximity Water Storage of Power Reactor Fuel, Summary Report, BAW-1484-7:" M. N. Baldwin, et al.; July, 1979, and (2) the Pacific Northwest Laboratory program "Criticality Experiments with Subcritical Clusters of 2.35 and 4.31 wt% U-Enriched UO_2 Rods in Water with Steel Reflecting

Walls," S. R. Bierman and E. D. Clayton: Nuclear Technology, Volume 54, page 131; August, 1981. More up-to-date information on material compositions for the latter experimental program was obtained from documentation in the International Handbook of Evaluated Criticality Safety Benchmark Experiments prepared by the Nuclear Energy Agency and the Organization for Economic Cooperation and Development.

The experiments from these two programs were selected for the benchmarking of the SCALE methodology employed in analyzing the St. Lucie Unit 2 fuel storage rack based on the following key parameters which span the variables of interest.

1) The fuel rods employed in the experiments contain sintered UO_2 and employ low absorption clad tubes;

2) The enrichments of the fuel rods employed in the critical experiments are 2.549 wt% for the B&W fuel and 4.306 wt% for the PNL fuel, thus providing a close approximation to the range of interest for the storage rack analyses;

3) The B&W experiments employ a 3 x 3 array of 14 x 14 fuel rod clusters and the PNL experiments employ a 2 x 2 array of rod clusters with typical rod cluster sizes of 9 x 12 or 11 x 14 rods;

4) Eight of the B&W experiments employ 0.188-inch thick SS-304 isolation sheets in the water channels separating the rod clusters in the 3 x 3 array and two of the PNL lattices employ either 0.191-inch or 0.119-inch thick SS-304 isolation sheets in the cross shaped channel separating the rod clusters in the 2 x 2 array;

5) The PNL experiments extrapolate to critical on number of fuel rods with non-borated moderator whereas the B&W experiments employ fixed numbers of fuel rods in each lattice and go critical on water height for a given soluble boron concentration.

Therefore, the geometry and composition of the critical experiments provides a good match to the St. Lucie Unit 2 spent fuel pool.

2.11 The usual allowance for possible uncertainties in the fuel depletion analysis is 0.01 delta-k at 30,000 MWD/MTU applied linearly (5% of the reactivity decrement to the burnup of interest). Justify your use of 0.005 delta-k.

FPL Response to Question 11:

The reactivity of the spent fuel pool as a function of burnup depends on the input nuclide concentrations as a function of burnup and on the microscopic cross sections assigned to these nuclides. The evolution of the nuclide concentrations was calculated by the two-dimensional multi-group transport code DIT, with an 89 group cross section library based on ENDF/B-VI. Benchmarking of the ROCS-DIT design codes using ENDF/B-VI cross sections for several operating cycles demonstrated a small reactivity bias as shown in Figure 1. The reactivity bias is defined as the calculated reactivity at the measured core critical conditions, as specified by the core exposure, critical boron concentration, power level, inlet temperature and control rod insertion. This reactivity bias is bounded by the bias of 0.005 delta-k per 30,000 MWD/T applied to the St. Lucie Unit 2 spent fuel pool. This lends confidence to the reactivity level and to the nuclide concentrations generated for spent fuel pool criticality analyses.

Two conservatisms were also included in the nuclide concentration determination for the St. Lucie Unit 2 spent fuel pool. The first conservatism lies in the soluble boron concentration used in the DIT depletions. The assembly DIT depletions were performed at a soluble boron concentration of 1000 ppm. The high boron concentration results in a hardened spectrum, magnifies the conversion ratio and results in an increased fissile content at discharge. Therefore, the assembly reactivity will be conservatively overestimated when placed in the spent fuel pool.

The second conservatism lies in the fuel temperature used in the DIT depletion. A constant temperature of 1200° F was assumed throughout the depletion, rather than the more realistic relationship where fuel temperature decreases with increasing burnup. The high fuel temperature increases the U-238 resonance absorption, increases the conversion ratio and magnifies the fissile content. From the above discussion, it can be concluded that the fissile content will contribute to a slight overprediction of the reactivity. A portion of these conservatisms have been used to address a reactivity effect associated with Pu-241 decay.

An additional component of the spent fuel pool reactivity lies in the cross sections used in KENO. KENO uses a 44 group cross section library based on ENDF/B-V. A comparison of DIT and KENO reactivities for the same set of nuclide concentrations was performed for a pin cell and a whole assembly. The most straightforward comparison between the DIT and KENO cross section libraries is through a pin cell reactivity calculation. This comparison was done for both a fresh fuel pin enriched to 4.5 wt%, and a fuel pin having an initial enrichment of 4.5 wt% depleted to 40,000 MWD/T.

Using the same set of nuclide concentrations, the KENO pin cell shows a reactivity gain over the DIT pin cell of 276 pcm by 40,000 MWD/T. The KENO whole assembly calculation includes a further conservatism by assuming that the pin burnup distribution is flat, i.e. there was no provision to account for the impact of the water holes on the burnup distribution. This assumption is conservative, because the fuel rods adjacent to the water holes exhibit a higher burnup and a lower k-infinity. These pins have a higher importance in the assembly because of the softer spectrum and higher flux induced by the large water holes. Therefore, an assembly modeled with a uniform burnup is more reactive than an assembly having a distributed burnup at 40,000 MWD/T.

The discussion provided above shows the conservatism of KENO relative to DIT. Having established the good reactivity performance of DIT as a function of burnup, one can conclude that KENO is conservative in its reactivity estimates for depleted fuel, and that the reactivity bias of 0.005 delta-k per 30,000 MWD/T is sufficient.

2.12 St. Lucie Unit 2 is requesting to take credit for soluble boron in the spent fuel pool (SFP) as a means to maintain k_{eff} less than 0.95. Standard technical specifications (TS) for Combustion Engineering designed plants do not credit soluble boron, yet still

include a surveillance requirement to sample the SFP every 7 days. Please explain why the surveillance should not be included in St. Lucie's TSs.

FPL Response to Question 12:

FPL has reviewed the Standard Technical Specifications Limiting Condition for Operation (LCO) 3.7.17, Fuel Storage Pool Boron Concentration, and its basis, which are provided in Volumes 1 and 3 of NUREG 1432. Based on our review of this LCO, 10 CFR 50.36, and Specification 5.6.1a. of St. Lucie Unit 2 TS, FPL believes that Specification 5.6.1a. and the plant procedures which implement the minimum boron concentration requirement provide assurance that the effective neutron multiplication factor of the spent fuel pool will not exceed 0.95. The rationale for this conclusion is presented below.

10 CFR 50.36(c) defines Surveillance Requirements to be: requirements relating to test, calibration or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within the safety limits, and that the limiting conditions for operation will be met.

FPL believes that the proposed limitations on fuel pool boron concentration do not meet the definition of a safety limit as given in 10 CFR 50.36(c)(1) because the fuel pool is not at risk to experience an uncontrolled release of radioactivity due to an unmonitored soluble boron concentration or with a boron concentration of 0 ppm. Postulated initiators of fission product release from the fuel pool typically involve cladding failure due to drop or impact.

10 CFR 50.36(c)(2) defines an LCO as the lowest functional capability or performance levels of equipment required for safe operation of the facility. Application of this definition to fuel pool soluble boron would result in a boron concentration value for surveillance purposes substantially less than the 1720 ppm value proposed by FPL. As noted in FPL's license submittal, a fuel pool boron concentration of as little as 520 ppm following any inadvertent dilution will still ensure keff remains less than 0.95.

In addition to the lower boron value that would satisfy the requirements for a Limiting Condition for Operation, NUREG 1432 LCO 3.7.17 suffers from an additional defect when used in the context of fuel pool soluble boron credit: fuel pool boron concentration is only required to be greater than the specified value and the surveillance is only required to be performed during the period between initiation of fuel movement and the completion of a fuel pool verification or inventory. While the period of applicability is sufficient to ensure that the fuel pool contains no misloaded fuel assemblies, this LCO will not provide ongoing assurance that the Unit 2 fuel pool keff is less than 0.95.

St. Lucie TS 5.6.1a. requires that the soluble boron concentration in the spent fuel pool be maintained greater than or equal to 1720 ppm and, unlike the Limiting Condition for Operation proposed by NUREG 1432, this requirement is applicable at all times. At St. Lucie, two procedures primarily implement the Specification 5.6.1a. boron requirement.

St. Lucie plant Chemistry Operating Procedure (COP) 05.04, Revision 1, Chemistry Department Surveillances and Parameters, Appendix B, "*Schedule for Periodic Tests*", requires that the boron concentration of the Unit 2 spent fuel pool shall be determined at least once per 7 days. Appendix D of this procedure, "*Nuclear Chemistry Parameters*", provides limits and the normal value range for the boron concentration in the spent fuel pool and the refueling water tank (RWT). This procedure uses a limit value of 1720 ppm and gives a normal value range of between 1800 and 2050 ppm.

In addition to the requirements noted above, COP-05.03, Revision 2, Refueling Shutdown/Startup Guidelines, provides guidance on sampling frequency for fuel pool boron during the different phases of Mode 6 operation. While in Mode 6 but prior to the initiation of fuel movement, boron concentration is sampled daily. During fuel movement and prior to the completion of core verification, fuel pool boron is sampled at least once per 12 hours. Following core verification, boron concentration is sampled each day for the remainder of Mode 6 operation. FPL believes increased sampling frequencies are appropriate during Mode 6 operation because of the interconnection of makeup flow and purification paths between the refueling cavity, the fuel transfer canal, and the spent fuel pool.

To ensure that these boron sampling frequencies and the normal range values given above are not modified without an evaluation of the consequences of any change, a reference to FPL's proposed license amendment evaluation and, if appropriate, to the NRC-generated safety evaluation, will be added to each procedure discussed above following the approval of FPL's license amendment submittal.

For these reasons, FPL believes that including the NUREG 1432 surveillance in St. Lucie's TS is not necessary. St. Lucie Unit 2 Design Features Specification 5.6.1a. and the existing implementing procedures ensure that the spent fuel pool boron concentration and the effective neutron multiplication factors remain bounded by analysis assumptions.

3.0 Conclusions

The responses to NRC-generated questions provided above render additional elaboration on areas addressed in the original PLA submittal prepared by FPL. Each of the responses support the conclusions of the PLA submittal and the no significant hazards determination prepared by FPL.

Figure 1

Reactivity Bias
Calculated Reactivity for Four Cycles with ENDF/B-VI