Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

August 8, 2011

L-2011-300
10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: St. Lucie Plant Unit 1
Docket No. 50-335
Renewed Facility Operating License No. DPR-67

Response to NRC Reactor Systems Branch Request for Additional Information Regarding Extended Power Uprate License Amendment Request

References:

(1) R. L. Anderson (FPL) to U.S. Nuclear Regulatory Commission (L-2010-259), "License Amendment Request for Extended Power Uprate," November 22, 2010, Accession No. ML103560419.
(2) Email from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 1 EPU draft RAIs (Reactor Systems – criticality)," dated May 27, 2011.
(3) Email from C. Wasik (FPL) to T. Orf (NRC), "St. Lucie Unit 1 Draft RAIs; SRXB 8-14," dated June 9, 2011.
(4) Email from T. Orf (NRC) to C. Wasik (FPL), "RE: St. Lucie Unit 1 Draft RAIs; SRXB 8-14," dated June 22, 2011.

By letter L-2010-259 dated November 22, 2010 [Reference 1], Florida Power & Light Company (FPL) requested to amend Renewed Facility Operating License No. DPR-67 and revise the St. Lucie Unit 1 Technical Specifications (TS). The proposed amendment will increase the unit's licensed core thermal power level from 2700 megawatts thermal (MWt) to 3020 MWt and revise the Renewed Facility Operating License and TS to support operation at this increased core thermal power level. This represents an approximate increase of 11.85% and is therefore considered an Extended Power Uprate (EPU).

By email from the NRC Project Manager dated May 27, 2011 [Reference 2], additional information related to the St. Lucie Unit 1 spent fuel pool criticality analysis was requested by NRC staff in the Reactor Systems Branch (SRXB) to support their review of the EPU LAR. The request for additional information (RAI) identified seven draft questions. In an email dated June 9, 2011 [Reference 3], FPL requested clarification of draft RAI's SRXB-11 and 12 as presented in Reference 2. In an email dated June 22, 2011 [Reference 4], the NRC provided clarification as requested in Reference 3 with respect to draft RAI SRXB-12 and stated that draft RAI SRXB-11 be disregarded. The response to the six remaining draft RAIs is provided in Attachment 1 to this letter.

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NRR

In accordance with 10 CFR 50.91(b)(1), a copy of this letter is being forwarded to the designated State of Florida official.

This submittal does not alter the significant hazards consideration or environmental assessment previously submitted by FPL letter L-2010-259 [Reference 1].

This submittal contains no new commitments and no revisions to existing commitments.

Should you have any questions regarding this submittal, please contact Mr. Christopher Wasik, St. Lucie Extended Power Uprate LAR Project Manager, at 772-429-7138.

I declare under penalty of perjury that the foregoing is true and correct to the best of my knowledge.

Executed on August 08, 2011.

Very truly yours,

Richard L. Anderson
Site Vice President
St. Lucie Plant

Attachment

cc: Mr. William Passetti, Florida Department of Health

Response to Request for Additional Information

The following information is provided by Florida Power & Light in response to the U. S. Nuclear Regulatory Commission's (NRC) Request for Additional Information (RAI). This information was requested to support Extended Power Uprate (EPU) License Amendment Request (LAR) for St. Lucie Nuclear Plant Unit 1 that was submitted to the NRC by FPL via letter (L-2010-259), dated November 22, 2010, Accession Number ML103560419.

In an email dated May 27, 2011 from NRC (Tracy Orf) to FPL (Chris Wasik), Subject: St. Lucie Unit 1 EPU draft RAIs (Reactor Systems – criticality)," the NRC requested additional information regarding FPL's request to implement the EPU. The RAI consisted of seven draft questions from the NRC's Reactor Systems Branch (SRXB). In an email dated June 9, 2011, Subject: "St. Lucie Unit 1 Draft RAIs; SRXB 8-14," FPL requested clarification of draft RAIs SRXB-11 and 12. In an email dated June 22, 2011, Subject: "St. Lucie 1 EPU draft RAIs (Reactor Systems – criticality)," the NRC provided the requested clarification with respect to draft RAI SRXB-12 and furthermore stated that RAI SRXB-11 be deleted. The six RAI questions and associated FPL responses are documented below.

SRXB-8
Identify and justify the fuel assembly design that was used for the NFV analysis.

Response

The NFV analysis was performed with the EPU fuel that was described in Section 2.8.1 of LAR Attachment 5. This is the same EPU fuel design that was used in the spent fuel pool analysis described in Section 2.8.6.2 of LAR Attachment 5. This is the only fuel assembly design to be stored in the NFV after the EPU.

SRXB-9
The optimum moderation case where the new fuel vault is subjected to low density moderation (e.g., fog or foam) was analyzed at 9 percent water density. Discuss and justify the applicability of the MCNP4a validation at these low density moderator conditions.

Response

In response to the NRC RAI regarding benchmark calculations for optimum moderation case, MCNP4a benchmark calculations have been updated with 8 new experiments. These 8 new experiments represent the most densely packed module and consist of a 15x17 matrix of fuel pins (less four corner pins) loaded on a triangular pitch. This module is referred to as triangular or T type. This T type experimental configuration results in an H/X of 17.43, where H/X is the atom ratio of moderator to fuel. The H/X for the optimum moderation condition (9% water density) for the new fuel vault is 10.04. The following represents a description of the 8 new experiments.

Serial Number	Reference	Identification	Enrichment	σ_{exp}	MCNP4a	H/X
1	B&W-1645	T-type, Core I, 435 ppm Boron	2.46	0.003	0.9982 ± 0.0007	17.43
2	B&W-1645	T-type, Core I, 426 ppm Boron	2.46	0.003	0.9961 ± 0.0006	17.43
3	B&W-1645	T-type, Core I, 406 ppm Boron	2.46	0.003	0.9971 ± 0.0005	17.43
4	B&W-1645	T-type, Core I, 383 ppm Boron	2.46	0.003	0.9983 ± 0.0006	17.43
5	B&W-1645	T-type, Core I, 354 ppm Boron	2.46	0.003	0.9980 ± 0.0007	17.43
6	B&W-1645	T-type, Core I, 335 ppm Boron	2.46	0.003	0.9960 ± 0.0006	17.43
7	B&W-1645	T-type, Core II, 361 ppm Boron	2.46	0.003	0.9944 ± 0.0006	17.43
8	B&W-1645	T-type, Core III, 121 ppm Boron	2.46	0.003	0.9907 ± 0.0006	17.43

The eight new experiments are added to the set of 56 current experiments. The set of 64 experiments provides the basis from which statistical analyses are performed. The guidance of NUREG/CR-6698 is used to perform the statistical analyses. The following table presents the bias and bias uncertainty.

Number of Experiments	Bias	Bias Uncertainty	Normality Test
64	-0.0015	0.0091	Passed

In order to satisfy the normality assumption, Pearson's chi-square ($\chi2$) test [Reference 1] is employed on the set of 64 experiments. Additionally, linear trend analysis is performed to determine whether there is any linear relationship between the MCNP calculated k-effs and H/X. Based on the coefficient of determination (R^2=1.75E-07), it is determined that there is no significant linear relationship between calculated k-effs and H/X [Reference 1]. Figure 1 presents the distribution of the k-eff values versus the trending parameter, which in this case is the H/X. This plot also visually confirms that there is no significant linear trend. Since there is no trend, extrapolation outside the H/X range, especially to 10.04 from 17.43 for the new fuel vault, is acceptable and therefore the bias and bias uncertainty calculated above are applicable for the new fuel vault calculations. Additionally, NUREG/CR-6698 states that (in Section 5) 10 percent extrapolation from the validation data (H/X range used is 17.43-398.69) is permissible.

The max k-eff value is reevaluated by using the new bias and bias uncertainty and is presented in the following table.

	Original	Revised with New Bias and Bias Uncertainty
Bias Uncertainty (95/95)	0.0090	0.0091
MCNP Uncertainty (95/95)	0.0008	0.0008
Tolerance	0.0027	0.0027
Total Uncertainty (95/95)	0.0094	0.0095
MCNP Bias	0.0012	0.0015
Calculated max k-eff	0.9661	0.9661
Max Total k-eff	0.9767	0.9771

The max total k-eff is increased by ~41 pcm, which is rather insignificant. Note that for all the new fuel vault calculations, the steel rack structure is conservatively replaced by water in the MCNP model.

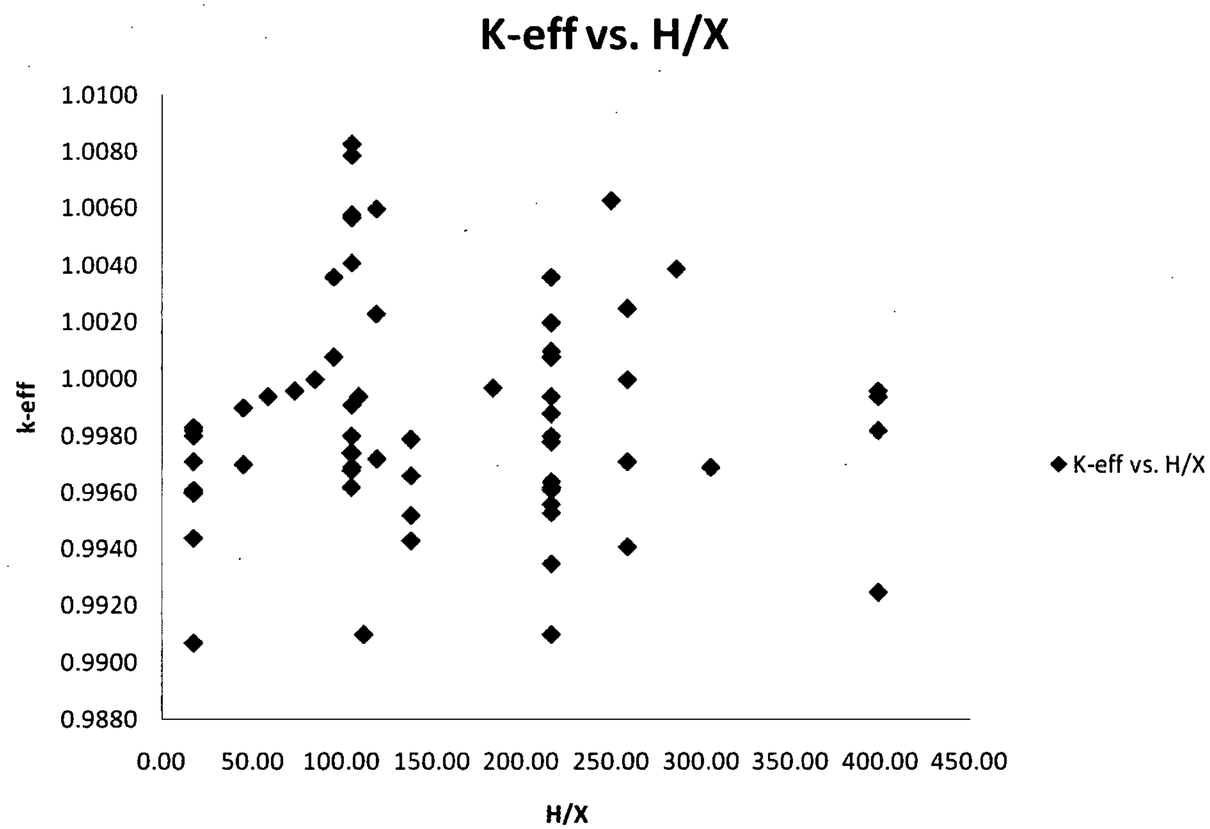

Figure 1: k-eff vs. H/X plot

References

[1] J.R. Taylor, An Introduction to Error Analysis (University Science Books, Mill Valley, California, 1982).

SRXB-10

Provide the letdown curve(s) used to justify the boron concentration used in the depletion calculations.

Response

The soluble boron concentrations used in the depletion calculations are 750 ppm for pre-EPU conditions, and 900 ppm for EPU conditions. These values are established using the letdown curves from past cycles (Cycles 19 to 23), the extended power uprate (EPU) first transition cycle (Cycle 24) and a representative EPU equilibrium cycle (Cycle N+2). The letdown curves for each of those cycles are provided in Tables 10-1 to 10-7.

The cycle average boron concentrations (C_{ave}) are calculated by the following equation:

$$C_{ave} = \frac{\sum_{i=1}^{n-1}[(Bu_{i+1} - Bu_i) * \frac{(C_{i+1} + C_i)}{2}]}{Bu_n - Bu_1}$$

Where:

n is the number of data in the boron letdown curve for each cycle,

Bu_i is the i_{th} burnup, and

C_i is the i_{th} boron concentration in the curve

The calculated cycle average boron concentration for each cycle is listed in Table 10-8. These results confirm that the value used for depletion for pre-EPU conditions conservatively bounds past operations, and the value used for depletion for EPU conditions has sufficient margin to cover future operation.

Table 10-1: St. Lucie Unit 1 Cycle 19 Boron Letdown

EFPH	Boron (ppm)
0	1423
100	1099
200	1082
500	1041
1000	995
2000	919
3000	839
4000	768
5000	704
6000	646
7000	580
8000	494
9000	395
10000	283
11000	163
11500	104
12000	46
12360	4

Table 10-2: St. Lucie Unit 1 Cycle 20 Boron Letdown

EFPH	Boron (ppm)
0	1303
100	984
200	969
500	929
1000	882
2000	805
3000	726
4000	655
5000	591
6000	532
7000	470
8000	392
9000	295
10000	183
10500	125
11100	53
11475	9
11520	4

Table 10-3: St. Lucie Unit 1 Cycle 21 Boron Letdown

EFPH	Boron (ppm)
0	1394
100	1070
200	1054
500	1015
1000	969
2000	895
3000	821
4000	755
5000	699
6000	647
7000	587
8000	498
9000	385
10000	263
11000	140
11400	92
11600	68
12000	21
12095	10

Table 10-4: St. Lucie Unit 1 Cycle 22 Boron Letdown

EFPH	Boron (ppm)
0	1347
100	1029
200	1013
500	971
1000	923
2000	841
3000	760
4000	687
5000	621
6000	560
7000	495
8000	407
9000	299
10000	178
11000	61
11388	16

Table 10-5: St. Lucie Unit 1 Cycle 23 Boron Letdown

EFPH	Boron (ppm)
0	1091
100	783
200	767
500	729
1000	685
2000	611
3000	537
4000	472
5000	416
6000	367
7000	328
8000	283
9000	207
9304	176
9804	123
10304	68
10892	1

Table 10-6: St. Lucie Unit 1 Cycle 24 Boron Letdown

EFPH	Boron (ppm)
0	1372
100	1038
200	1016
500	976
1000	938
2000	876
3000	815
4000	766
5000	722
6000	673
7000	602
8000	500
9000	372
10000	235
11000	102
11714	10

Table 10-7: St. Lucie Unit 1 Cycle N+2 Boron Letdown

EFPH	Boron (ppm)
0	1488
100	1148
200	1129
500	1086
1000	1039
2000	954
3000	866
4000	788
5000	716
6000	648
7000	584
8000	506
9000	396
10000	264
11000	130
12000	0

Table 10-8: Average Boron Concentration for Each Cycle

	Cycle	Average Boron Concentration (ppm)
Pre-EPU	19	597
	20	529
	21	597
	22	557
	23	405
EPU	24	608
	N+2	623

Note: Analysis boron concentration value for depletion is 750 ppm for pre-EPU and 900 ppm for EPU.

SRXB-11

This RAI has been deleted as per Email from T. Orf (NRC) to C. Wasik (FPL), "St. Lucie 1 EPU draft RAIs (Reactor Systems – criticality)," dated June 22, 2011.

SRXB-12

Demonstrate that the presence of vessel flux reduction assemblies (VFRAs) during depletion do not lead to higher reactivity of the surrounding assemblies as compared to depletion without VFRA.

Response

Vessel Flux Reduction Assemblies (VFRAs) contain full length fuel rods with depleted uranium at an axially constant initial enrichment of approximately 0.3 wt%. During operation, these assemblies had hafnium absorber rods in the guide tube locations.

A total of sixteen (16) VFRAs were used during four cycles of operation from 1991 through 1997; eight were initially loaded in Cycle 11 and reloaded for Cycle 12, and a different set of eight were initially loaded in Cycle 13 and reloaded for Cycle 14. All sixteen VFRAs were loaded on corners of the core periphery flats ("dog ears") for both cycles of operation. Note that each corner of the core periphery flats only has one face adjacent location (also on the core periphery), and two locations on the semi-periphery that are "half" face-adjacent. The figure below depicts the core locations for the VFRAs.

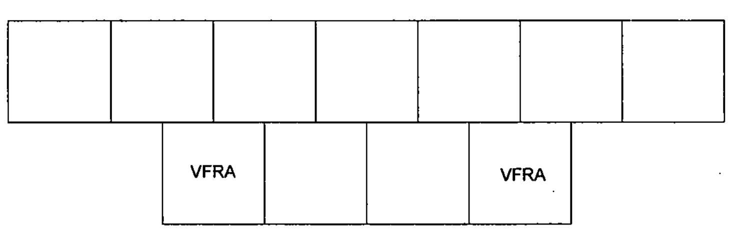

Figure 1 – VFRAs Core Locations

The VFRAs operated at very low power levels due to three factors – (1) their low initial enrichment, (2) hafnium absorber rods, and (3) located on the core periphery. Typically, the VFRAs had relative powers less than 50% of typical peripheral assemblies.

Core designs for cycles that included VFRAs took the local power decrease into consideration to ensure successful designs and operation with the peaking factors within the design limits. For example, more reactive assemblies or fresh assemblies would be placed in core locations near the VFRAs than would be typically done in the core semi-periphery, in order to counteract the power decrease from the VFRAs. The impact of the VFRAs on the limiting peaking factors for the core was negligible.

With regards to spectral effects caused by the presence of VFRAs and potential impact on neighboring fuel assemblies, no deviations in core response was observed during the cycles with VFRAs. Note that there were only a total of 96 neighboring fuel assemblies (16 VFRAs X 2 cycles X 3 neighboring locations).

Generic studies [References 1, and 2] indicate that one of the more dominant operating parameters that results in higher reactivities is the moderator temperature used for depletion. As discussed in the criticality report HI-2104714 Revision 1 Section 7.4, the moderator temperature used for depletion is taken as the peak power assembly exit temperature for all cycles of operation. Recall that the VFRA neighboring assemblies are on the periphery or semi-periphery, thus operated at low power for at least one cycle of operation. Therefore, any potential reactivity impact caused by the VFRAs would be bounded by the moderator temperature used for depletion in the criticality analysis.

References:

[1] "Assessment of Reactivity Margin and Loading Curves for PWR Burnup-Credit Cask Designs", NUREG/CR-6800 / ORNL/TM 2002/6, March 2003.

[2] M. D. DeHart, "Sensitivity and Parametric Evaluations of Significant Aspects of Burnup Credit for PWR Spent Fuel Packages", ORNL/TM-12973, May 1996.

SRXB-13

Describe the surveillance program on the control element assemblies (CEAs) credited to ensure the required subcritical margin.

Response

The current spent fuel pool (SFP) criticality analysis, approved in License Amendment 193 regarding soluble boron credit, credited control element assemblies (CEAs) in some storage configurations. During the NRC review of that license amendment, FPL responded to two questions related to the acceptability of CEAs for use in reactivity control in the SFP criticality analysis. Those responses are documented in References 1 (Question 11) and 2 (Question 2), and acknowledged by the NRC in Reference 3 (Safety Evaluation page 7). The acceptability of CEAs for reactivity control in the SFP storage racks is established by analyzing CEA conditions up to the time of discharge into the SFP. FPL has performed a unit-specific evaluation of the environmental factors affecting CEAs during reactor operation, including fretting wear, fast neutron flux, clad strain, and poison depletion. Specifically, poison depletion is minimized while a CEA is in the core because CEAs are normally fully withdrawn from the active fuel region during power operation. The evaluation of these factors is discussed in the responses provided to the NRC in References 1 and 2.

From the evaluation of the environmental factors affecting CEA in-reactor performance and results of an earlier measurement and inspection campaign of CEAs placed in the SFP, conservative values of CEA in-reactor lifetime have been established. Configuration controls are established at the plant to ensure that CEA use is maintained within the established lifetimes. Once discharged into the spent fuel pool, the environmental factors observed in-reactor are no longer present or are significantly reduced. Thus, the effect of the pool environment on the CEA life or its performance as an absorber is insignificant. The Inconel-625 cladding used in the CEA rods is known to withstand the typical spent fuel pool chemical environment with no detrimental effects.

Additionally, plant procedures for inspecting new CEAs and testing CEAs throughout their in-reactor life for acceptable insertion times would identify physical abnormalities that could potentially impact the use of CEAs for reactivity control in the SFP. Any CEAs found with cracked cladding will be held in reserve and not credited in the spent fuel pool storage array, unless an inspection campaign or subsequent analysis deems them acceptable for use.

Based on the: a) analytical methods used to establish CEA lifetimes, b) operation in the reactor within established lifetimes, c) initial receipt inspection and d) subsequent testing of insertion times every cycle during the in-reactor lifetime, it is concluded that the CEAs credited for SFP use will be intact and within their design basis to perform their reactivity control function.

References:

[1] FPL Letter L-2003-125, "St. Lucie Unit 1 – Docket No. 50-335, RAI Response for Spent Fuel Pool Soluble Boron Credit," May 14, 2003 (Question 11). ML031390240

[2] FPL Letter L-2003-245, "St. Lucie Unit 1 – Docket No. 50-335, RAI Response for Proposed Amendment - Spent Fuel Pool Soluble Boron Credit," September 29, 2003 (Question 2). ML032740110

[3] NRC Letter, Brendan T. Moroney to J.A. Stall, "St. Lucie Plant, Unit 1 – Issuance of Amendment Regarding Spent Fuel Pool Soluble Boron Credit (TAC No. MB6864)," September 23, 2004. ML042670562

SRXB-14

Provide the nominal areal densities of the Boral and METAMIC absorbers at SL1. Provide the applicable standard deviations of the areal densities of the Boral and METAMIC absorbers at SL1.

Response

The B-10 areal densities for the two materials are defined as follows:

Boral: 0.030 ± 0.002 g/cm^2, i.e. 0.028 g/cm^2 minimum

Metamic: 0.016 g/cm^2 nominal, 0.015 g/cm^2 minimum

Note that the manufacturing variations for those materials are specified as a tolerance (i.e. where all material is required to be within the specified range), or directly as a minimum value, but not as a standard deviation (i.e. where a specific percentage of the material would be required to be within the specified range).

Further note that the design basis calculations only use the minimum values listed above.