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THOMPSON,H.L. Division of Licensing

SUBJECT: Application to amend License DPR-67,permitting continued operation at rated thermal power for specified period of time following dropped control element assembly.No significant hazards evaluation encl.

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P. O. BOX 14000, JUNO BEACH, FL 33408

FLORIDA POWER & LIGHT COMPANY

JUL 19 1985

L-85-268

Office of Nuclear Reactor Regulation
Attention: Mr. Hugh L. Thompson
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Thompson:

Re: St. Lucie Unit No. 1
Docket No. 50-335
Proposed License Amendment
Movable Control Assemblies

In accordance with 10 CFR 50.90, Florida Power & Light Company submits herewith three signed originals and forty copies of a request to amend Appendix A of Facility Operating Licenses DPR-67.

The requested change will permit continued operation at Rated Thermal Power for a specified period of time following a dropped Control Element Assembly. Furthermore, the current action statement C will be reformulated into new action statements C and H. This reformulation will better correlate the requirements for corrective action in a Technical Specification with the underlying analytical assumptions the action statements are designed to protect.

A no significant hazards evaluation has been performed as required by 10 CFR 50.91 and 92 and is provided.

The proposed amendment has been reviewed by the St. Lucie Plant Facility Review Group and the Florida Power & Light Company Nuclear Review Board.

In accordance with 10 CFR 50.91(b)(1), a copy of the proposed amendment is being forwarded to the state designee for the State of Florida.

In accordance with 10 CFR 170.21, a check is attached as remittance for the license amendment application fee for St. Lucie Unit 1.

A001 Recd w/att Check
1/1

Should you have any questions regarding this submittal, please feel free to contact us.

Very truly yours,

J. W. Williams, Jr.
Group Vice President
Nuclear Energy

JWW/RG/cab

Attachments

cc: Dr. J. Nelson Grace, Region II

Harold F. Reis, Esquire

Lyle E. Jerret, Ph.D, Director
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SAFETY EVALUATION AND DETERMINATION OF NO SIGNIFICANT HAZARDS MODIFICATION TO ST. LUCIE 1 TECHNICAL SPECIFICATION 3/4.1.3

Introduction:

Two changes have been proposed for St. Lucie Unit 1 Technical Specification 3/4.1.3. The purpose of the first change is to permit Unit 1 to continue to operate at rated thermal power for some period of time following an inadvertent single dropped control element assembly CEA. The intent of the second change is to reformulate an existing Action statement (Action C) into two separate action statements (Actions C and H) to more clearly link any required operator action with the applicable analysis assumptions requiring that action.

The first proposed change will permit St. Lucie Unit 1 to continue to operate at rated thermal power for a period of time following an inadvertent single dropped CEA . This period of time will depend on the pre-drop value of the integrated radial peaking factor (F_R) measured at the plant during normal power distribution surveillances. The only transient affected by this proposed Technical Specification change is the single CEA drop. The CEA drop accident is defined as the electrical or mechanical failure of the CEA drive mechanism which opens the circuit of the holding coil causing the CEA to drop into the core. The single CEA drop transient is an important part of determining the plant DNB-related operating space.

Initially, the CEA drop event causes a decrease in the reactor power. The heat extraction by the secondary plant remains essentially constant however, causing the average reactor coolant temperature to decrease. This temperature decrease, combined with the assumed end of cycle negative value of the moderator temperature coefficient, will cause the reactor power level to return to its initial power level with the dropped CEA still remaining in the core. The presence of the dropped CEA will result in a distorted core power distribution and increased peaking factors.

In plants such as St. Lucie Unit 1 with analog-type Reactor Protective Systems, there is no need for a specific trip signal or other automatic action to be generated following an inadvertent dropped CEA. Instead, sufficient margin has been designed into the plant operating space, specifically in the DNB Limiting Condition for Operation (LCO), to ensure acceptable consequences for the worst dropped CEA at any time during core life. Additionally, this design margin is complemented by the action of the Reactor Protective System to inhibit automatic CEA withdrawal during a CEA drop event. This feature has been credited in the CEA drop analysis.

For St. Lucie Unit 1, margin was designed into the DNB LCO through the input values chosen for the XCOBRA (Reference 1) thermal margin analysis model. A 10% greater input value (1.87) of F_R, after uncertainties, than the Technical Specification limit of F_R (1.70) was used. Even using the greater input value of F_R (1.87) in the thermal margin analysis, the resulting DNBR's were greater than the DNB Specified Acceptable Fuel Design Limits (SAFDL). The margin between the permissible normal operation limit of 1.70 (or actual lower measured value) and the 1.87 thermal margin input value can be utilized as available overpower margin (AOPM) for the single CEA drop analysis. Table 1 details the specific cycle 5 & 6 single CEA drop results.

From analysis results the increase in assembly peak F_R values following a dropped CEA event was seen to be a function of the reactivity worth of the dropped CEA and the assembly's distance from the dropped CEA. Because of this an assembly other than the one with the core maximum F_R can have a larger percent increase than the core maximum F_R assembly. For cycles 5 and 6 the maximum F_R increase in a non-peripheral assembly was calculated to be 9.2% of its initial (pre-drop) value immediately following the dropped rod event and 11.7% after one hour. Peripheral assemblies contained the greatest percentage F_R increase. However, these assemblies are of low power and are not limiting. They were not considered in selecting the maximum F_R increase.

As can be seen from the attached tabulated data, for both cycles 5 and 6 the increase in the maximum core-wide value of F_R one hour following a CEA drop is less than 10%. This means that for cycles 5 and 6 if the before-drop F_R was equal to 1.70, one hour following the CEA drop the maximum F_R would have increased 8.3% to approximately 1.84. This value is less than the 1.87 value used as input to generate the DNB LCO, therefore, the plant could remain at 100% power for one hour following the worst case CEA drop at any time during cycles 5 or 6. To assure the CEA drop results from future cycles will be bounded, Figure 3.1-1a in the proposed Technical Specification was drawn to permit only 15 minutes of full power operation when the pre-drop value of F_R equals 1.70. As the pre-drop value of F_R decreases below 1.70, the time St. Lucie Unit 1 may remain at full power after a drop increases up to a maximum of one hour as can be seen in Figure 3.1-1a. From a reactor operation standpoint, values of $F_R \geq 1.67$ are not anticipated to occur. To further assure Figure 3.1-1a remains bounding, the increase in the core maximum F_R for the CEA drop transient will be analyzed for each future cycle.

As stated above, the proposed Technical Specification change attached requires the misaligned CEA be realigned with the rest of its bank within a specified amount of time depending on the pre-drop measured F_R. If the CEA cannot be realigned within this time period, reactor power must be reduced to $\leq$ 70% of rated power. Within the time constraints given in Figure 3.1-1a,

the analysis presented in this report demonstrates that the peaking factor increase during the one hour period will not exceed that utilized in the safety analysis for the dropped CEA event.

The second proposed change to Specification 3/4.1.3 consists of the reformulation of Action Statement C into two Action Statements, C and H. This change will better correlate the requirements for corrective action in a Technical Specification with the underlying analytical assumptions the action statements are designed to protect. The reason for this action statement is to assure that the assumptions made in the safety analysis regarding the core power distribution (specifically axial shape analysis) during the cycle depletion bound the power distributions seen in the core during actual operation. These assumed power distributior are used in several plant safety analysis and are also used in generating the Unit 1 operating setpoints. Validity of these assumptions can be assured by limiting, as is done in Specification 3.1.3.6, the time duration operation may continue with CEAs inserted beyond the Long Term Insertion Limits (LTIL). Specification 3.1.3.6 limits this insertion to less than or equal to 14 EFPD per year. This time limit, which is applicable to Specification 3/4.1.3, will ensure the power distribution as actually depleted in the core closely approximates an unrodded power distribution depletion. If operation beyond the LTIL was permitted at rated thermal power in excess of 14 days per calendar year, the resulting cycle power distribution would begin to significantly deviate from the unrodded distribution assumed. Analysis of this condition could require the modification of plant transient analysis.

When CEAs are positioned within their alignment requirements and at a withdrawn position greater than the LTIL as is covered by Action C of Specification 3/4.1.3, then the resulting power and burnup distributions will remain bounded by the power distributior used for plant transient and setpoint analysis independent of the length of time the CEAs remain inserted. This is because the overall perturbation of the power distribution from the ARO power shape due to this amount of CEA insertion is small.

As noted above, St. Lucie Unit 1 proposes to recognize the distinctions in safety analysis requirements outlined above by reconstructing the present action statement into two different action statements; one with applicability when CEAs are above the LTIL and a separate one when CEAs are inserted beyond the LTIL. This separation will aid operations personnel to better understand the underlying technical basis of each specification and action statement and it will aid in the standardization of specifications between St. Lucie Units 1 and 2. No changes in safety analysis results or input are required as a result of this separation or the addition of Figure 3.1-1a. Therefore, as required by 10CFR50.92(c)(1), the proposed changes to Specificatio 3/4.1.3 do not result in an increase in the probability or consequenc of any accident previously evaluated because no change in analysis input or assumptions was required for any transient. Acceptable results will continue to be shown for all previously analyzed transients.

The proposed changes to St. Lucie Technical Specification 3/4.1.3 do not create the possibility of new or different type of accident from any accident previously evaluated because neither the configuration of the plant nor its mode of operation have been modified. Because no changes will be made to the physical plant or its mode of operation as a result of this Technical Specification change, there is no increase in the possibility of a new or different type of accident as discussed in 10CFR50.92(c)(2).

The proposed changes to the St. Lucie 1 CEA Position Technical Specification will not result in any reduction in the margin of safety as discussed in 10CFR50.92(c)(3) because no inputs to nor results from plant safety analysis require change or modifications. The required overpower margin for each transient analyzed for St. Lucie 1 is completely unaffected by this proposed change therefore, the difference between reactor safety limits and the results of the safety analysis, which is representative of the margin of safety, is unchanged.

Based on the information presented above, Florida Power & Light Company has concluded that the proposed change to the St. Lucie Unit 1 Technical Specifications does not constitute an unreviewed safety issue or a significant hazard to the health and safety of the public as discussed in 10CFR50.92(c).

Reference 1: N.F. Fausz and T.W. Patten, XCOBRA-IIIC: A Computer Code to Determine the Distribution of Coolant During Steady State and Transient Core Operation XN-NF-21(P), Rev. 2, September 1982.

TABLE 1

ST. LUCIE UNIT 1

INCREASE IN F_R VERSUS TIME FOR CYCLE 5 AND 6 CEA DROP ANALYSIS

Increase in F_R (%)

	BOC5	EOC5	BOC6	EOC6
a)	4.8%	6.0%	6.3%	6.5%
b)	6.8%	7.2%	7.5%	8.3%
c)	9.2%	7.7%	8.3%	8.0%
d)	11.7%	9.7%	10.6%	10.2%

a) Increase in Core Maximum F_R immediately following CEA drop

b) Increase in Core Maximum F_R 1 hour following CEA drop

c) Maximum increase in F_R anywhere within the core immediately following drop

d) Maximum increase in F_R anywhere in the core 1 hour following CEA drop