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AUTH.NAME AUTHOR AFFILIATION
WILLIAMS,J.W. Florida Power & Light Co.
RECIP.NAME RECIPIENT AFFILIATION
EISENHUT,D.G. Division of Licensing

SUBJECT: Application for amend to License NPF-16,revising Tech Specs to reflect changes required to commence Cycle 2 operation. Affidavit,reload safety evaluation summary & "Reload Safety Rept" encl.App I to rept withheld (ref 10CFR2.790). 566 Prop C/Phase 5 Packet

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FLORIDA POWER & LIGHT COMPANY

June 4, 1984

L-84-148

Office of Nuclear Reactor Regulation
Attention: Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Eisenhut:

Re: St. Lucie Unit No. 2
Docket No. 50-389
Proposed License Amendment
Cycle 2 Reload

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 License NPF-16.

This amendment is submitted to reflect changes required to commence operation of Cycle 2, which is currently scheduled for November 25, 1984. Therefore, NRC approval is requested on or before November 25, 1984.

The proposed changes are summarized in the attached St. Lucie Unit 2 Cycle 2 Reload Safety Evaluation Summary, and are shown on the accompanying marked-up Technical Specification pages. A detailed Reload Safety Report is attached.

It should be noted that the proposed changes permit operation of St. Lucie Unit 2 Cycle 2 at the licensed power level of 2560 MWt. However, the analyses incorporate and bound operation for core power levels up to 2700 MWt. Authorization for operation up to 2700 MWt. will be requested in a future license amendment application.

In accordance with 10 CFR 50.91(a)(1), it has been determined that the proposed amendment does not involve any significant hazards considerations pursuant to 10 CFR 50.92. The No Significant Hazards Considerations determination is attached.

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.

Appendix I to the attached Reload Safety Evaluation Report is proprietary information, and therefore, exempt from public disclosure in accordance with 10 CFR 2.790.

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

The proposed amendment has been determined to be a Class IV amendment. A check for $12,300.00 is attached in accordance with 10 CFR 170.22.

Very truly yours,

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

JWW/RJS/db

Attachment

cc: J. P. O'Reilly
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
101 Marietta Street, N.W., Suite 2900
Atlanta, GA 30303

Lyle Jerrett, Ph.D., Director
Office of Radiation Control
Dept. Health & Rehabilitative Services
1317 Winewood Boulevard
Tallahassee, FL 32301

AFFIDAVIT PURSUANT

TO 10 CFR 2.790

Combustion Engineering, Inc.)
State of Connecticut)
County of Hartford) SS.:

I, A. E. Scherer, depose and say that I am the Director, Nuclear Licensing, of Combustion Engineering, Inc., duly authorized to make this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.790 of the Commission's regulations and in conjunction with the application of Florida Power and Light Company for withholding this information.

The information for which proprietary treatment is sought is contained in the following document:

Statistical Combination of Uncertainties - FP&L Unit 2, Cycle 2 Reload Report Appendix I .

This document has been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Combustion Engineering in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.790 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1. The information sought to be withheld from public disclosure are the methodology related to the determination of the probability distributions for specific uncertainties and the combination of uncertainties to be used in determining plant setpoints and related technical specifications, which is owned and has been held in confidence by Combustion Engineering.

2. The information consists of test data or other similar data concerning a process, method or component, the application of which results in a substantial competitive advantage to Combustion Engineering.

3. The information is of a type customarily held in confidence by Combustion Engineering and not customarily disclosed to the public. Combustion Engineering has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The details of the aforementioned system were provided to the Nuclear Regulatory Commission via letter DP-537 from F.M. Stern to Frank Schroeder dated December 2, 1974. This system was applied in determining that the subject document herein are proprietary.

4. The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.790 with the understanding that it is to be received in confidence by the Commission.

5. The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

6. Public disclosure of the information is likely to cause substantial harm to the competitive position of Combustion Engineering because:

a. A similar product is manufactured and sold by major pressurized water reactor competitors of Combustion Engineering.

b. Development of this information by C-E required tens of thousands of man-hours and hundreds of thousands of dollars. To the best of my knowledge and belief a competitor would have to undergo similar expense in generating equivalent information.

c. In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of methods to statistically combine uncertainties and determine uncertainty probability distributions for specific uncertainties.

d. The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e. The information consists of methods and statistical models used to combine uncertainties and the resultant net uncertainty to be applied in determining plant setpoints and technical specifications, the application of which provides a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Combustion Engineering, take marketing or other actions to improve their product's position or impair the position of Combustion Engineering's product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

f. In pricing Combustion Engineering's products and services, significant research, development, engineering, analytical, manufacturing, licensing, quality assurance and other costs and expenses must be included. The ability of Combustion Engineering's competitors to utilize such information

without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

g. Use of the information by competitors in the international marketplace would increase their ability to market nuclear steam supply systems by reducing the costs associated with their technology development. In addition, disclosure would have an adverse economic impact on Combustion Engineering's potential for obtaining or maintaining foreign licensees.

Further the deponent sayeth not.

A. E. Scherer
Director
Nuclear Licensing

Sworn to before me
this 18th day of May

Lydia A. Smith
Notary Public

LYDIA A. SMITH, NOTARY PUBLIC
STATE OF CONNECTICUT No. 69542
COMMISSION EXPIRES MARCH 31, 1989

STATE OF FLORIDA)
) ss.
COUNTY OF DADE)

J. W. Williams, Jr., being duly sworn, deposes and says:

That he is a Group Vice President 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.

Appendix I to the Reload Safety Report is proprietary, and therefore, exempt from public disclosure in accordance with Section 2.790 of the NRC "Rules of Practice", Title 10, Code of Federal Regulations.

J. W. Williams, Jr.

Subscribed and sworn to before me this

4th day of June, 19 84

NOTARY PUBLIC, in and for the County of Dade, State of Florida.

My commission expires July 20, 1987

St. Lucie Unit 2 Cycle 2
Reload Safety Evaluation
Summary

This report provides a safety evaluation for the operation of St. Lucie Unit 2 Cycle 2 at 2560 MWt. Technical Specification changes are required to enable operation with 18 month cycles and low leakage fuel management configurations. The report provides the necessary analysis to support these Technical Specification changes. The analysis incorporates a change to the CEA (control element assembly) configuration to obtain greater flexibility in operational control and a reduction in the minimum required reactor coolant (RCS) flow to gain sufficient margin between measurable flow and required flow. In addition, the analysis incorporates and bounds operation with a core power level of up to 2700 MWt (although a request to increase the rated core power to 2700 MWt is not included). The required analysis for 2700 MWt also includes a recalculation of containment pressure and temperature during transients. A request for authorization for operation up to 2700 MWt will be submitted in a future license amendment application.

The safety evaluation makes use of the Statistical Combination of Uncertainties (SCU) methodology in the analysis to provide for a more realistic assessment of system instrumentation uncertainties, system processing uncertainties, manufacturing tolerances and modeling uncertainties. This methodology together with several proposed Technical Specification changes, which are more restrictive than Cycle 1, provide the extra margin to accommodate more economical fuel management designs, a reduced required minimum RCS flow, and a core power level of up to 2700 MWt without a significant increase in the consequences of potential accidents or any reduction in safety margin (i.e., all required safety criteria are met).

The safety evaluation generally follows the NRC Standard Review Plan (SRP) guidelines in the performance of the safety analyses, with any deviations sufficiently justified. The analyses do not employ any new or unreviewed methodology (the SCU methodology was previously used and NRC approved for St. Lucie Unit 1, Calvert Cliffs Unit 2 and Arkansas Nuclear One Unit 2).

The proposed Technical Specification changes are summarized in the attached table. These proposed Technical Specification changes and the supporting safety evaluation has been reviewed by the St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board and found to be necessary and meet all the required safety criteria.

TABLE 4-1

ST. LUCIE UNIT 2
TECHNICAL SPECIFICATION AND BASES CHANGES

Change No.	Page	Specification	Action	Remarks
	2-1	2.1.1.2	Change peak linear heat to centerline melt limit from 21.0 kw/ft to 22.0 kw/ft	Peak linear heat to centerline melt limit is raised to the calculated limit for Cycle 2, as described in Section 2.2.**
	2-3	Figure 2.1-1	Replace this figure with revised figure.	Thermal limit lines are being changed to reflect analysis at 2700 MWT, Technical Specification radial peaking factors and the implementation of margin recovery programs.
	2-4	Table 2.2-1	The Containment Pressure - High Trip: Allowable value is being reduced from 5.0 psig to 4.1 psig.	This change is being made so that the trip setpoint is consistent with the assumptions made to the containment pressure High - High trip setpoint in the LOCA containment pressure and the pre-trip steam line inside containment analyses, Section 3.3.4.
	2-5	Table 2.2-1	Change design reactor coolant flow from 370,000 gpm to 363,000 gpm on Footnote (*).	All analyses sensitive to minimum flow requirements were performed assuming a 363,000 gpm minimum guaranteed flow rate.
	2-9	Figure 2.2-3	Replace figure with revised figure.	The TM/LP LSSS is being changed to reflect analysis at 2700 MWT, Technical Specification radial peaking factors, and the implementation of margin recovery programs.
	2-10	Figure 2.2-4	Replace figure with revised figure.	The TM/LP LSSS is being changed to reflect analysis at 2700 MWT, Technical Specification radial peaking factors, and the implementation of margin recovery programs.

**Refers to sections contained in Reload Safety Report.

Change No.	Page	Specification	Action	Remarks
	B2-1	B2.1.1	Change minimum DNBR limit from "1.20" to read "an acceptable limit".	The value of DNBR, which corresponds to the 95/95 criteria, changes slightly from cycle to cycle due to the application of statistical uncertainty analysis; specific values of the DNBR limit are being deleted to avoid the necessity of cycle-by-cycle Tech. Spec. Revisions.
	B2-1 B2-4	B2.1.1 B2.2.1	Change statement on DNBR from "1.20" to "the acceptable minimum DNBR limit".	The value of DNBR, which corresponds to the 95/95 criteria, changes slightly from cycle to cycle due to the application of statistical uncertainty analysis; specific values of the DNBR limit are being deleted to avoid the necessity of cycle-by-cycle Tech. Spec. revisions.
	B2-2	Figure B2.1-1	Replace figure with revised figure.	Figure is being changed to reflect higher radial peaking.
	3/4 1-3	3/4.1.1.2.	Change shutdown margin for Mode 5 from 2.0% delta k/k to 3.0% delta k/k.	The shutdown margin is being increased to reflect the assumptions used in the boron dilution event, Section 3.2.4.4.
	3/4 1-8 3/4 1-10 3/4 1-12	3.1.2.2 3.1.2.4 3.1.2.6	Change shutdown margin from 2.0% delta k/k to 3.0% delta k/k.	To be consistent with Technical Specification 3/4.1.1.2.
	3/4 1-14	3.1.2.8	Change shutdown margin at 200° from 2.0% delta k/k to 3.0% delta k/k.	To be consistent with Technical Specification 3/4.1.1.2.

Change No.	Page	Specification	Action	Remarks
	3/4 1-18 3/4 1-19 3/4 1-19a	3.1.3.1	Reduce number of CEA regulating groups from 6 to 5 in Items b.2 and h. of the Action Statement.	Due to change in number of CEA regulating banks.
			Reword Item d. to reflect use of figure showing dropped CEA recovery time vs. measured F_r^T.	
			Remove Footnote (#) which showed the time constraints on a single CEA drop. This is now contained in Item d. which includes a figure showing dropped CEA recovery time vs. measured F_r^T.	Change to reflect higher radial peaks used in analysis to support increased dropped CEA recovery time flexibility.
			Resequence Items e. through g. to reflect addition of new Item e.	
	3/4 1-24	3.1.3.4	Change CEA drop time from 3.0 seconds to 2.7 seconds.	This reduced time is consistent with plant measurements.
	3/4 1-28	Figure 3.1-2	Replace figure with revised figure.	The PDIL is being changed to accommodate the new CEA rod pattern.
	3/4 2-4	Figure 3.2-2	Replace figure with revised figure.	LHR Ex-core LCO is being revised to reflect analysis at 2700 MWt, Technical Specification radial peaking factors, and the implementation of margin recovery programs.
	3/4 2-5	Figure 3.2-3	Replace figure with revised figure.	Allowable combinations of thermal power and F_r^T, Fxy^T are being revised to reflect analysis at 2700 MWt and the implementation of margin recovery programs.

Change No.	Page	Specification	Action	Remarks
	3/4 2-7	3.3.2	Change the F^T_{xy} limit from 1.60 to 1.75.	The value for Fxy^T limit is raised to reflect the value used in the safety analysis.
	3/4 2-9	3.2.3	Change the F^T_r limit from 1.60 to 1.70.	The value for F^T_r limit is raised to reflect the value used in the safety analysis.
	3/4 2-9 3/4 2-11	4.2.3.2 Table 3.2-1	Delete all references to rod bow penalty.	Rod bow penalties have been accommodated in the revised DNBR limit of 1.28.
	3/4 2-12	Figure 3.2-4	Replace figure with revised figure.	The DNB-LCO is being changed to reflect analysis at 2700 MWt, Reactor Coolant Flow of 363,000 gpm, Technical Specification radial peaking factors, and the implementation of margin recovery programs.
	3/4 2-15	Table 3.2-2	Increase upper bound of cold leg temperature from 548°F to 549°F.	Upper bound cold leg temperature change reflects safety analysis assumptions performed for Cycle 2.
			Decrease reactor coolant flow rate from 370,000 gpm to 363,000 gpm.	All analyses sensitive to minimum flow requirements were performed assuming a 363,000 gpm minimum guaranteed flow rate.
	3/4 3-6	Table 3.3-2	Change Containment Pressure - High response time from 1.55 seconds to 1.15 seconds.	This reduced time is consistent with plant measurements.
	3/4 3-17	Table 3.3-4	Changed containment spray on Containment Pressure High - High Trip Setpoint from 9.30 psig to 5.40 psig and the allowable value from 9.40 psig to 5.50 psig.	This change was made to be consistent with assumptions in the containment pressure analysis.
			Change the Containment Pressure High Trip Setpoint from 5.0 psig to 4.7 psig and the allowable value from 5.10 psig to 4.80 psig.	This change was necessary because of the change made to the Containment Pressure High - High Trip Set Point.

Change No.	Page	Specification	Action	Remarks
	3/4 3-20	Table 3.3-5	Change Feedwater Isolation Response Time from ≤5.35/5.35 to ≤5.15/5.15 for both Containment Pressure - High and Steam Generator Pressure - Low.	This change is being made to incorporate the specified valve closing time and to eliminate the 0.25 second additional conservatism that was assumed in Cycle 1.
	3/4 4-9	3.4.3	Change minimum and maximum pressurizer indicated level from 65% to 68.0%.	This change is being made to be consistent with a new pressurizer level program and assumptions made in the excess charging event, Section 3.2.5.1.
	3/4 7-1 3/4 7-2 3/4 7-3	3/4.7.1 Table 3.7-1 Table 3.7-2	Replace these pages with revised pages.	Changes made to allowable power values reflect analysis at 2700 MWt. Format of specification has been changed to improve clarity.
	3/7 7-10	3.7.1.6	Change full closure times of 5.6 seconds and 5.35 seconds both to 5.15 seconds.	These changes reflect appropriate closure times for the main feedwater isolation valve (5.15 seconds was assumed in peak containment pressure analysis.)
	B3/4 1-1 B3/4 1-2	B3/4.1.1.1 & B3/4.1.1.2 B3/4.1.2	Change the required shutdown margin with Tavg ≤ 200° from 2.0% delta k/k to 3.0% delta k/k.	The shutdown margin is being increased to reflect the assumptions used in the boron dilution event, Section 3.2.4.4.

Change No.	Page	Specification	Action	Remarks
	B3/4 1-4	B3/4.1.3	Remove wording indicating at what power levels a DNBR SAFDL violation could occur, and clarify the wording on how this potential violation is eliminated. Increase steady-state radial peak from $F_r^T = 1.60$ to $F_r^T = 1.70$.	Change wording, since power levels at which a DNBR SAFDL violation may occur could vary slightly from cycle to cycle.
	B3/4 1-4	B3/4.1.3	Change actual radial peak for additional margin from $F_r^T = 1.50$ to $F_r^T < 1.70$. Change Item 5 from a 30 minute misalignment time for an $F_R^T \leq 1.50_T$ to 60 minutes for an $F_r^T \leq 1.55$.	These changes reflect the assumptions utilized in the single drop CEA analysis found in Section 3.2.4.3.
	B3/4 2-2 B3/4 2-3	B3/4.2.2, B3/4.2.3 & B3/4.2.4 Table B3/4.2-1	Delete last paragraph which discusses rod bow penalties, and delete table on rod bow penalties.	Rod bow penalties have been accommodated in revised DNB limit of 1.28.
	B3/4 2-2	B3/4.2.5	Change "minimum DNBR limit of $\geq$ 1.20" to "an acceptable minimum DNBR".	The value of DNBR, which corresponds to the 95/95 criteria, changes slightly from cycle to cycle due to the application of statistical uncertainty analysis; specific values of the DNBR limit are being deleted to avoid the necessity of cycle-by-cycle Tech. Spec. revisions.

Change No.	Page	Specification	Action	Remarks
	B3/4 7-1	B3/4.7.1.1	Replace page with revised page.	Changes made to allowable power values reflect analysis at 2700 MWt. Format of specification has been charged to improve clarity.
	5-3	5.3.1	Change "...236 fuel rods clad..." to "...236 fuel and poison rod locations. All fuel and poison rods are clad..."	This new statement is appropriate if assemblies with poison rods are loaded into the core. Cycle 2 will contain such assemblies.
			Change "...a maximum total weight of 1698.5 grams uranium" to "...approximately 1700 grams uranium".	The weight of 1698.5 grams is a Cycle 1 maximum weight. By wording it approximately 1700 grams, variations in loading weights can be tolerated.
	5-1	5.2.1	Change containment net free volume from 2.5×10^6 ft^3 to 2.506×10^6 ft^3.	Change in this value represents a more detailed analysis of the containment net free volume.
	5-3	5.3.2	Increase the number of full-length control element assemblies (CEAs) from 83 to 91.	Eight full length CEAs are being added into vacant part length CEA locations.

NO SIGNIFICANT HAZARDS CONSIDERATIONS

ST. LUCIE 2 CYCLE 2

OPERATION AT 2560 MWTH

DOCKET: 50-389
LICENSE: NPF-16

MAY 1984

I. INTRODUCTION

The requested amendment to the St. Lucie Unit 2 operating license is being submitted in support of the upcoming Cycle 2 core reload. The requested amendment will incorporate technical specification changes as discussed in the evaluation. The reload will involve replacing approximately one-third of the reactor core and additional new Control Element Assemblies will be installed in existing equipped locations. The Region D fresh fuel assemblies to be used in this reload are not significantly different from those previously found acceptable to the NRC for St. Lucie Unit 2 Cycle 1. The analytical methods used to demonstrate conformance with the technical specifications and regulations have been previously approved by the NRC staff. In addition, the proposed technical specification changes do not change the applicable acceptance criteria previously approved by the NRC Staff. The evaluation performed in support of this amendment has determined that, when measured against the standards in 10CFR50.92, no significant hazards consideration exists. It is also concluded that this amendment involves no unreviewed safety questions per 10CFR50.59.

II. TECHNICAL SUMMARY

The St. Lucie Unit 2 nuclear power plant is presently licensed to operate at a rated thermal power of 2560 Mwth with a physical configuration as defined and described by the FSAR. This reload involves removing depleted fuel assemblies from approximately one-third of the nuclear core and replacing them with fresh fuel of a similar type as previously loaded. The maximum nominal enrichment of the Region D fresh fuel will be 3.65 weight percent uranium - 235 as compared to a nominal maximum enrichment in Cycle 1 of 2.73 w/o. The fresh fuel assemblies will also incorporate minor dimensional changes as a result of design changes recognized as desirable at other C-E plants. These changes create a larger space between the top of each fuel rod and the fuel upper end fitting flow plate thus allowing greater space for fuel rod expansion. The fuel assembly guide tubes will be changed from cold worked zircaloy to annealed zircaloy which will result in a lower growth rate of the fuel assembly. The increase in enrichment is incorporated in the Region D fuel assemblies to provide for an extended fuel cycle length. There has been no change to the fuel design bases and as such the new fuel continues to satisfy General Design Criteria 10 and 11 and other design bases considered in the Staff review of the fuel for Cycle 1.

III. BASIS FOR PROPOSED NO SIGNIFICANT HAZARDS CONSIDERATION

An evaluation of this request for amendment has been performed to demonstrate that no significant hazards consideration exists, based upon a comparison with the criteria of 10CFR50.92(c). The requested technical specification changes have been categorized into several subheadings for the purposes of this evaluation.

A. CHANGES TO SAFETY LIMITS

Refinements in calculational techniques have led to the following two proposed changes.

1. The minimum value of the DNBR during steady-state operation, normal operational transients and anticipated transients is increased from 1.20 to 1.28.

2. The allowable limit on peak linear heat rate of the fuel is increased from 21 kw/ft to 22 kw/ft.

There has been no change to the criteria used to establish these safety limits. The proposed DNBR value still provides at least a 95% probability at a 95% confidence level that Departure from Nucleate Boiling (DNB) does not occur on a fuel rod having that minimum DNBR during steady state operation or during anticipated operational occurrences. The evaluation of the various factors associated with DNB will now be based on the Statistical Combination of Uncertainties (SCU) methodology (Appendix I of the Reload Safety Report). This methodology also incorporates adjustments for rod bow directly in the DNB limit, whereas in the reference cycle (Cycle 1) rod bow was accounted for explicitly in the monitoring of the radial peaking factor. The SCU methodology is described in C-E report CEN-123(F)-P, and has been previously reviewed and approved by the NRC. Application of the techniques to the plant specific parameters of St. Lucie Unit 2 is described in the accompanying Reload Safety Report.

The proposed new value for peak linear heat rate is still a value corresponding to centerline fuel melt as determined by the fuel evaluation model, FATES-3. The power-to-centerline melt limit for Cycle 2 takes credit for decreased power peaking which is characteristic of highly burned fuel. Also, since a decrease in fuel melt temperature accompanies burnup, the most limiting power-to-centerline melt has been found to occur at an intermediate burnup range. Using conservative estimates of the burnup point at which the power peaking begins to decrease and the rate at which it decreases for Cycle 2, the most limiting power-to-centerline melt has been determined to be in excess of 22 kw/ft.

These revised safety limits have been factored into the safety analyses performed for this reload application and all results are within previously established criteria and design basis; hence, no reduction in safety margin has resulted from these changes.

These technical specifications provide a numerical value with which to judge and verify the acceptability of safety analyses that are performed. Therefore, these changes have no impact on accident probability and consequence, for either accidents previously analyzed or the potential for different accidents.

Therefore, these proposed changes may be considered similar to the example in 10CFR50.92 for amendments that are considered not likely to involve significant hazards considerations:

> "(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

B. TECHNICAL SPECIFICATION CHANGES TO ENHANCE OPERATING MARGIN

The allowed plant operating space as defined and controlled by the technical specifications is revised in the following areas.

1. The allowable planar radial peaking factor (F^T_{xy}) has been increased from 1.60 to 1.75 and the allowable integrated radial peaking factor (F^T_r) has been increased from 1.60 to 1.70.

2. The minimum required Reactor Coolant System (RCS) flow has been reduced from 370,000 gpm to 363,000 gpm.

3. The maximum allowed cold leg temperature has been increased from 548 F to 549 F.

4. Increased restrictions to the LSSS and LCOs are implemented to offset the effects of the increased operating space produced by items 1, 2 and 3.

Detailed calculations were performed to evaluate the impact of these changes on Anticipated Operational Occurences and Postulated Accidents. The extent of these analyses can be characterized within the following six categories.

1. Increase in heat removal by the secondary system. (Section 3.2.1)

2. Decrease in heat removal by the secondary system. (Section 3.2.2)

3. Decrease in reactor coolant flowrate. (Section 3.2.3)

4. Reactivity and power distribution anomalies. (Section 3.2.4)

5. Decrease in reactor coolant system inventory. (Section 3.2.6)

6. Loss of Coolant events. (Section 3.3)

NOTE: Section numbers refer to the sections in the Reload Safety Report.

The criteria for judging the acceptability of these events has not changed from the reference cycle (Cycle 1). The detailed results of these calculations are provided in the accompanying Reload Safety Report along with comparisons with the appropriate limiting criteria. The following discussion provides a summary of various events analyzed with respect to the three basic criteria; i.e., offsite dose, reactor coolant system pressure, and fuel performance.

1. <u>Offsite Dose</u>

Acceptance guidelines for offsite radiation dose continue to be based on 10CFR100 criteria. The most limiting postulated accident with respect to offsite dose was determined to be a steamline break outside of containment (Section 3.2.1.5b). The detailed analysis of this postulated accident includes assumptions such as concurrent loss of AC power and the most adverse values for the process parameters (RCS temperature, pressure, core MTC, NSSS power, etc.) that affect the outcome of this event. Even with the conservatism assumed, the results are well within the limits of 10CFR100. The consequences of a steamline break inside containment are

even less severe with respect to offsite dose since the releases are confined within the containment building.

The limiting Anticipated Operational Occurence which is analyzed for impact on offsite dose is the Inadvertent Opening of a Steam Generator Safety Valve (Section 3.2.1.4). It is assumed that this event will result in a complete blowdown of one steam generator and partial blowdown of the other. Conservative assumptions to maximize the calculated doses include maximum steam generator and RCS radionuclide concentrations. The results continue to be a small fraction of 10CFR100 limits.

2. Reactor Coolant System Pressure

Acceptance guidelines for RCS pressure are based on RCS design limits as defined by General Design Criteria 14 and 15. The most limiting postulated accident with respect to RCS pressure was found to be a feedwater system pipebreak (Section 3.2.2.6). This event is analyzed with conservative assumptions, such as loss of AC power and the most adverse values for the process parameters that affect the results. Also, a parametric evaluation is performed to identify the exact break size that maximizes the RCS pressure peak. These conservative calculations show that the pressure peak resulting from this event is still below the RCS upset pressure limit of 2750 psia.

The limiting Anticipated Operational Occurence which affects RCS pressure is the Loss of Condenser Vacuum event (Section 3.2.2.3). The resulting loss of load causes an increase in steam generator pressure which is relieved by opening of the secondary safety valves. There is also an increase in RCS pressure which allows protective systems to initiate a reactor trip at the high pressure setpoint to terminate the event. The peak RCS pressure attained is well below the upset pressure limit of 2750 psia.

3. Fuel Performance

Criteria in this category require that a coolable fuel geometry is maintained such that continued removal of decay heat is ensured. This condition is met by maintaining fuel temperatures below the Specified Acceptable Fuel Design Limit (SAFDL) and limiting the duration of DNB during postulated accidents. The most limiting postulated accident with respect to fuel integrity was determined to be the Steamline Break Outside of Containment (Section 3.2.1.5b).

Note that this event has been previously discussed as the most limiting postulated accident with respect to offsite dose. The result indicates that only a small number of fuel pins are predicted to fail and a coolable geometry is maintained.

The limiting Anticipated Operational Occurrence that is considered in this category is the Total Loss of Forced RCS Flow (Section 3.2.3.2). The conditions assumed in this analysis include the maximum allowed cold leg temperature, maximum radial peaking factors and minimum RCS flowrate as proposed. A parametric analysis is performed to determine the axial shape index within the allowable range that provides the most severe results. This event is used to establish the minimum initial margin that must be maintained by the Limiting Conditions for Operation (LCOs) with respect to the DNBR limit. Hence, this event results in an acceptable minimum DNBR of 1.28.

Another set of criteria that is established to evaluate fuel performance is described by 10CFR50.46. Assurance that these criteria are satisfied is provided by the detailed analyses performed for small break LOCA, large break LOCA and post-LOCA long term cooling. The highest Peak Clad Temperature (PCT) calculated, resulted from a Double-Ended Guillotine Break at Pump Discharge (DEG/PD) with a PCT of 2041°F as compared to an allowable limit of 2200°F. A detailed description of these analyses and corresponding results is provided in the Reload Safety Report (Section 3.3.1). In all cases, the analytical results show acceptability with respect to the 10CFR50.46 criteria.

These detailed calculations show that incorporation of the increased operating space, when offset by the more limiting restrictions imposed by changes to the LSSS and LCOs result in limiting events which are still below the corresponding acceptance criteria. Therefore, no reduction in safety margin has occurred.

The combined results of these calculations when compared to the reference cycle (Cycle 1) show that these proposed changes do not result in any increase in the probability of those events previously analyzed and no significant increase in the consequences of these events can be shown.

None of these proposed changes result in any modifications to plant equipment; the minor variations in plant parameters are accounted for in the evaluations of AOOs and postulated

accidents as described above. Therefore, this evaluation has further concluded that these changes do not provide a potential for accidents different from those previously considered.

Since these proposed changes yield results which are well within acceptance criteria, the changes can be considered similar to the example provided in 10CFR50.92 for amendments that are considered not likely to involve significant hazards considerations:

> "(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

C. INCREASED CYCLE LENGTH

The mode 5 shutdown margin is increased from 2% to 3% delta k/k as a result of the fuel management program which will permit an increased cycle length.

The accompanying Reload Safety Report analyzes anticipated operational occurrences that are affected by the proposed changes in cycle length and Mode 5 shutdown margin. The limiting events with respect to radiological release and loss of shutdown margin are the following.

1. Inadvertent opening of a steam generator safety valve (Section 3.2.1.4).

2. Post trip analysis of a steam line break from Hot Full Power (Section 3.2.1.5.c).

3. Chemical Volume and Control System (CVCS) malfunction (Section 3.2.4.4).

These analyses were performed with bounding values of shutdown margin, rod worth, and boron worth for the current fuel loading. The results from an analysis of the inadvertent opening of a

steam generator safety valve show that reliable control of reactivity is maintained and that radiological doses at the site boundary are a small fraction of the 10CFR100 guidelines. The steam line break analysis shows that, with the same HZP shutdown requirement as for the previous cycle, there will be no significant return to power. Analysis of the CVCS malfunction (boron dilution) shows that under all operating and refueling conditions the time from annunciation to criticality will meet or exceed the required minimum criteria. Thus, all criticality criteria are met. Increases in fuel temperatures and coolant pressures are regulated by the constraints imposed by the LSSS and LCOs.

From these analyses it can be concluded that there is no significant increase in the probability and consequences of accidents previously analyzed. Nor do these changes create the possibility of a new or different kind of accident. The changes do not reduce the safety margin inasmuch as the safety analyses show that acceptable results are obtained with the same criteria pertaining to offsite dose rates, return to power and time from annunciation to criticality.

This change can be considered as being similar to the example in 10CFR50.92 for amendments that are considered not likely to involve significant hazards considerations:

> "(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications; for example, a more stringent surveillance requirement."

D. CEA RELATED CHANGES

Cycle 2 will incorporate several changes related to the Control Element Assemblies (CEAs), primarily to enhance operational characteristics such as control of axial shape index. A description of the physical configuration of these changes is provided here, along with a summary of the affected technical specifications.

Eight additional CEAs will be installed in core locations which are presently unrodded. These locations are already provided with drive motors, position indicating instrumentation and all associated hardware. This change will take advantage of these equipped locations to increase the total number of CEAs available for reactor control. These additional CEAs will also result in an increase in the available shutdown margin. The sequence in

which these eight new CEAs and the existing 83 CEAs are maneuvered will be changed. The 83 CEAs in the reference cycle (Cycle 1) are subdivided into six regulating and two shutdown banks. The 91 CEAs available for Cycle 2 will be subdivided into five regulating and two shutdown banks. This grouping change will increase the number of CEAs from four to twelve in the first sequentially inserted group during reactor control maneuvers. Also, the CEA insertion limitation (Power Dependent Insertion Limit, PDIL) will be revised. The changes to group configuration and PDIL will increase the amount of control available to plant operators and will allow for a more even application of CEA worth, which will minimize the effects on core radial power distribution.

Safety analyses have been performed to verify the acceptability of increasing the amount of time allowed to recover a dropped CEA. Plant experience with operational surveillance has shown that the actual CEA drop time associated with a reactor trip is conservatively faster than previously assumed for the reference cycle. Therefore, changes concerning CEA recovery time and CEA drop time will be incorporated into the technical specifications.

Detailed analyses of Anticipated Operational Occurrences which were performed to confirm the acceptability of these changes include the following:

1. Uncontrolled CEA withdrawal from a subcritical or low power condition (Section 3.2.4.1).

2. Uncontrolled CEA withdrawal at power (Section 3.2.4.2).

3. CEA misoperation (rod drop) (Section 3.2.4.3).

Analysis of these events have shown that there is no significant increase in the consequences of these events resulting from the proposed changes. The postulated accident which is most significantly affected by these proposed changes is the CEA Ejection Event. This event would result from the highly unlikely failure of a pressure housing which retains a CEA. The analysis of this event is performed in accordance with the NRC approved C-E methodology described by CENPD-190A (Section 3.2.4.6). The analysis shows that the most severe results, which occur at a zero power initial condition, predict that no fuel failures will occur. Therefore, acceptance criteria related to fuel performance and offsite dose are satisfied and no reduction in safety margin has resulted from these changes.

These changes are similar to the example in 10CFR50.92 for amendments that are considered not likely to involve significant hazards considerations:

> "(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

The physical implementation of these changes will be accomplished by modifications to the existing Control Element Drive Mechanism Control System (CEDMCS). Originally defined functional requirements and specifications for this equipment will be retained and, hence, there will be no impact on the probability of previously analyzed events and no potential for new events.

E. CONTAINMENT INTEGRITY

To assure containment integrity, the following changes are proposed:

1. Containment spray high-high trip setpoint is lowered from 9.30 psig to 5.40 psig and allowable values from 9.40 psig to 5.50 psig.

2. The high containment pressure setpoint for Engineered Safety Features (ESF) functions is lowered from 5.0 psig to 4.7 psig. The allowable value is reduced from 5.1 psig to 4.8 psig. The high containment pressure setpoint of 4.0 psig for reactor trip remains the same as Cycle 1, however, the allowable value is reduced from 5.0 psig to 4.1 psig.

3. The allowable response time for high containment pressure instrumentation is reduced from 1.55 seconds to 1.15 seconds.

These changes are similar to the example in 10CFR50.92 for amendments that are considered not likely to involve significant hazards considerations:

"(ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications: for example a more stringent surveillance requirement."

The lower containment spray trip setpoint results in lower peak containment pressure following mass and energy releases to the containment. The high containment pressure setpoints have been reduced as a result of the high-high containment pressure setpoint changes, to assure proper sequencing of automatic safety system actions. The reduction in response time is justified based on in-plant experience with instrument performance. Section 3.3.4 of the Reload Safety Report shows that with the proposed changes, a higher core power (2700 MWth) can be accommodated without compromising containment integrity. The report presents analyses that show peak containment pressures for a large break LOCA or a main steam line break, the two limiting transients for containment pressure, will be below the design pressure of 44 psig. Thus, the probability and consequences of previously analyzed events have not increased nor has the safety margin decreased. The probability for a new accident has not increased as no new failure mechansim has been introduced. The lower limit on initial containment pressure has not been changed, thereby assuring that the assumptions used in the ECCS analysis remain valid.

F. PRESSURIZER WATER LEVEL

A change to the pressurizer water level control system is incorporated to raise the normal operating water level in the pressurizer. This level program improvement will provide greater margin between the pressurizer heater cutoff level setpoint and the projected minimum water level following a reactor trip. Consequently, to accommodate this control system setpoint change, the maximum allowable indicated pressurizer water level is increased from 65% to 68%. This change has been accounted for in analysis of a CVCS malfunction (Section 3.2.5.1) which is the limiting event affected by this change. The analysis concludes that the operator has 20 minutes available to take corrective action following annunciation of the high pressurizer water level alarm to prevent filling the pressurizer. This is a sufficient and acceptable period of time for the operator to terminate the charging-letdown flow imbalance and hence no reduction in safety margin has occurred. This change also has no affect on the probability or consequence of new or previously analyzed accidents.

This increase in allowable pressurizer water level is similar to the example in 10CFR50.92 for amendments that are considered not likely to involve significant hazards considerations:

> "(vi) A change which either may result in some increase to the probability or consequences of a previously-analyzed accident or may reduce in some way a safety margin, but where the results of the change are clearly within all acceptable criteria with respect to the system or component specified in the Standard Review Plan; for example, a change resulting from the application of a small refinement of a previously used calculational model or design method."

G. SECONDARY SAFETY VALVE

The main steamline safety valve operability requirement is changed to incorporate revised maximum allowable power limits to be in effect when fewer than all safety valves are in service. This specification will now be of the same format and technical content as the corresponding St. Lucie Unit 1 requirement. The same calculational methods used for the reference cycle (Cycle 1) are applied here and no increase or decrease in rated valve capacity is assumed. The analyses which support this change are now based on steam flowrates which would be present with the plant operating at 2700MWth. The revised specification continues to comply with the ASME Boiler and Pressure Vessel Section III code requirements to limit peak secondary system pressure to 110% of design pressure. Therefore, no reduction in safety margin has occurred, and the probability/consequence of accidents is not affected.

Since this change results in a reduction in the allowed fractional power level, the change may be considered similar to the example in 10CFR50.92 for amendments that are considered not likely to involve significant hazards considerations:

> (ii) A change that constitutes an additional limitation, restriction, or control not presently included in the technical specifications; for example a more stringent surveillance requirement."

H. CORRECTIONS AND ADMINISTRATIVE CHANGES

The following two changes constitute editorial corrections in the existing technical specifications:

1. Section 5.3.1 - change "fuel rods" to "fuel and poison rods" to include fuel assemblies containing poison rods.
2. Section 5.3.1 - change "1698.3 grams" to "approximately 1700 grams" to permit minor variations in core loading and weight.

These changes are of an administrative nature and follow the example given in 10CFR50.92 for amendments that are considered not likley to involve significant hazards considerations:

> "(i) A purely administrative change to technical specifications: for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature."

IV. CONCLUSION

From the considerations detailed above it can be concluded that the proposed amendments to the St. Lucie Unit 2 Technical Specifications do not

a) increase the probability or consequences of accidents previously analyzed

b) increase the potential for accidents different from any accident previously considered

c) reduce the safety margin.

Therefore it is concluded that in accordance with the provisions of 10CFR50.92 the changes involve no significant hazards considerations.