SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 8
TO FACILITY OPERATING LICENSE NO. NPF-16
FLORIDA POWER AND LIGHT COMPANY, ET AL.
ST. LUCIE UNIT 2
DOCKET NO. 50-389

1.0 INTRODUCTION

By letter dated June 4, 1984, the Florida Power and Light Company (FP&L) submitted a request to reload and operate St. Lucie Plant Unit No. 2 for Cycle 2 (Ref. 1). In support of the request, the licensee submitted a reload safety analysis report (Ref. 2) and a statistical combination of uncertainties (SCU) methodology report (Ref. 3) applicable to St. Lucie 2.

The staff has reviewed the application and the supporting documents and has prepared the following evaluation of the fuel design, nuclear design, and thermal-hydraulic design of the core as well as an evaluation of those plant transients that were reanalyzed for Cycle 2. In addition, a summary and an evaluation of the Technical Specification changes reviewed are also presented.

Although the analyses incorporate and bound operation for core power levels up to 2700 MWt, this evaluation approves continued operation of St. Lucie 2 during its second fuel cycle at a power rating of 2560 MWt, the same core power level approved and licensed for the initial fuel cycle operation. Approval for operation at 2700 MWt would require an additional application for license amendment which we understand will be submitted in the near future.

2.0 FUEL DESIGN

2.1 Mechanical Design

The Cycle 2 core consists of 137 Batch B and C fuel assemblies irradiated during the first cycle in addition to 80 fresh Batch D assemblies. Except for the design features listed below, the mechanical design of the Batch D assemblies is identical to that of the Cycle 1 fuel assemblies. These refinements were made for the purpose of increasing margins for shoulder gap change and fuel assembly growth.

1. The fuel rod overall length has been reduced by 0.3 inches by shortening the plenum length. This results in additional shoulder gap clearance. The analysis of fuel rod internal pressure due to the shorter plenum length was performed with the fuel performance code, FATES3 (Refs. 4 and 5), which has been approved by the staff (Ref. 6). The calculations were performed assuming a larger rod plenum reduction than will occur for Cycle 2 and using conservatively high radial peaking factors versus pin burnup. The results indicate that the internal rod pressure will remain below the system pressure of 2250 psia for burnups up to 60,000 MWD/MTU. Therefore, the staff concludes that the effect of the shorter plenum length on Batch D rod internal pressure satisfies the NRC fuel rod pressure criterion.

2. The fuel assembly guide tube has been changed from cold worked to annealed material. This will result in a lower growth rate of the fuel assembly and is, therefore, acceptable.

3. The guide tube overall length has been increased by 0.4 inches. This produces a corresponding raising of the upper end fitting that results in additional shoulder gap clearance. Although the longer annealed guide tube may begin operation with a higher spring loading on the fuel assembly, the lower growth rate for annealed guide tubes will minimize the change in spring compressive force with increasing burnup. This change is, therefore, acceptable.

The licensee has stated that the cladding creep collapse time for any fuel that will be irradiated during Cycle 2 was conservatively determined to be greater than its maximum projected residence time. The creep collapse analysis was performed by Combustion Engineering (CE) using the CEPAN computer code (Ref. 7) which has been approved for licensing applications. The staff concludes that cladding collapse has been appropriately considered and will not occur for Cycle 2 operation.

Cycle 2 will consist of 73 Batch B assemblies and 64 Batch C assemblies. All Batch C assemblies and 16 Batch B assemblies have been shimmed to increase the initial shoulder gap clearance from 0.997 inches to 1.447 inches. The licensee has concluded that this increase is sufficient to assure at least 95% confidence of adequate shoulder gap clearance during Cycle 2 operation. This conclusion was based on Arkansas Nuclear One Unit 2 (ANO-2) measured shoulder gap closure in conjuction with predicted fluences to evaluate shoulder gap. Cycle 2 will also include 57 Batch B unshimmed assemblies with an initial shoulder gap of 0.997 inches. During the Cycle 1-2 outage, verification of an adequate shoulder gap for a second cycle of operation for these assemblies will take place by conducting shoulder gap measurements in conjunction with supporting analysis. Those assemblies that fail to show adequate shoulder gap for the Cycle 2 operation will be shimmed at the site. A formal report addressing this will be submitted to the NRC prior to Cycle 2 startup, as required by the St. Lucie 2 license condition on axial growth.

2.2 Thermal Design

The thermal performance of Cycle 2 fuel was performed by the licensee by analyzing a composite fuel pin that envelopes the various fuel assemblies (fuel Batches B, C, and D) in the Cycle 2 core using FATES3. The NRC-imposed grain size restriction (Ref. 6) was included and a power history that envelopes the power and burnup levels representative of the peak pin at each burnup interval from beginning-of-cycle (BOC) to end-of-cycle (EOC) was used. The power-to-centerline melt limit, determined by FATES3, takes credit for decreased power peaking that is characteristic of highly burned fuel. Since a decrease in fuel melt temperature accompanies burnup, the most limiting power-to-centerline melt was 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. Since approved methods and acceptable assumptions were used and this value has been incorporated into the proposed Technical Specifications and used in the safety analyses, the staff finds that the power-to-centerline melt limit of 22 kW/ft is acceptable.

3.0 NUCLEAR DESIGN

3.1 Fuel Management

The St. Lucie 2 Cycle 2 core consists of 217 fuel assemblies, each having a 16x16 fuel rod array. All of the 73 Batch A assemblies and 7 Batch B assemblies initially loaded in Cycle 1 will be removed and*replaced by 24 Batch B assemblies (3.65 w/o U-235 enrichment), 16 Batch D assemblies (3.65 w/o U-235 enrichment) containing 4 burnable poison shims per assembly, and 40 Batch D/ assemblies (3.65 w/o U-235 enrichment) containing 8 burnable poison shims per assembly. The revised high density fuel storage racks as well as the fresh fuel storage racks have been approved for storage of fuel of maximum U-235 enrichment of 4.5 weight percent.

The Cycle 2 core will utilize a low leakage fuel management scheme to reduce the uranium requirements for a specified total energy output. This is achieved by the loading of several once-irradiated Batch B and C assemblies on the core periphery and the inboard loading of most of the fresh Batch D assemblies. This scheme has been approved for many recent reload cores and has been accounted for in the calculation of the Cycle 2 core physics parameters. It is, therefore, acceptable.

The nuclear design and safety analysis for Cycle 2 is based on a Cycle 1 length of between 8,250 to 10,000 effective full power hours (EFPH). The analyses presented by the licensee will accommodate a Cycle 2 length up to 10,000 EFPH at a core power of 2700 MWt. This evaluation, however, approves continued operation of Unit 2 during its second cycle at a power rating of 2560 MWt, the same level approved and licensed for the initial fuel cycle operation.

3.2 Power Distributions

Hot full power (HFP) fuel assembly relative power densities are given in Reference 2 for beginning-of-cycle (BOC), middle-of-cycle (MOC), and end-of-cycle (EOC) conditions and for unrodded and rodded (CEA Bank 5 in) configurations. These results show that the Technical Specification limits on radial peaking factors bound the values expected to occur throughout the entire cycle. These expected values are based on three-dimensional ROCS code coarse-mesh and two-dimensional PDQ code fine-mesh core depletion calculations that have been approved previously by the NRC staff and are, therefore, acceptable.

3.3 Control Requirements

The value of the most restrictive required shutdown margin is determined by the EOC hot zero power (HZP) steam line break analysis and the resulting uncontrolled reactor coolant system (RCS) cooldown. A minimum shutdown margin of 5.0% k/k is required to control the reactivity transient. Based on this value of required shutdown margin and on calculated available scram reactivity including a maximum worth stuck control element assembly (CEA) and appropriate calculational uncertainties, sufficient excess exists between available and required scram reactivity to meet the Technical Specification limiting condition for operation (LCO). For operating temperatures below 200°F, the reactivity transients resulting from any postulated accident are minimal and a 3% k/k shutdown margin has been found to provide adequate protection. These results are derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

The CEA configuration for Cycle 2 differs from that of the reference cycle in several respects. These changes were made primarily to enchance operational characteristics such as control of axial shape index (ASI) and will also result in an increase in the available shutdown margin. Eight additional CEAs will be installed in the empty part length CEA drives since Cycle 2 contains no part length rods. The CEA banks and subgroups have been reconfigured and a new lead bank has been installed consisting of 12 reduced strength CEAs. Each of these consist of two B_4C fingers and three Al_2O_3 fingers. This will increase the number of CEAs from 4 to 12 in the first sequentially inserted group during reactor control maneuvers. The 91 CEAs available will now be subdivided into five regulating and two shutdown banks.

The effects of these CEA configuration changes have been properly accounted for in the safety analyses and in the Technical Specifications and have been derived using approved methods. Therefore, the staff finds the changes acceptable.

4.0 THERMAL-HYDRAULIC DESIGN

Steady-state thermal hydraulic analysis for Cycle 2 is performed using the approved core thermal hydraulic code TORC and the CE-1 critical heat flux (CHF) correlation. The core and hot channel are modeled with the approved method described in Ref. 8. The design thermal margin analysis is performed using the fast running variation of the TORC code, CETOP-D (Ref. 9). In response to the staff's request, the licensee has shown that the CETOP-D model predicts minimum DNBR conservatively relative to TORC (Ref. 10).

The uncertainties associated with the system parameters are combined statistically using the approved statistical combination of uncertainties (SCU) methodology described in Refs. 11, 12, and 13. Using this SCU methodology, the engineering hot channel factors for heat flux, heat input, rod pitch and cladding diameter are combined statistically with other uncertainty factors to arrive at an equivalent DNBR limit of 1.28 at a 95/95 probability/confidence limit. It has been calculated using the approved method described in Ref. 14. The value used for this analysis, 1.75% MDNBR, is valid for bundle burnups up to 30,000 MWD/MTU. For those asemblies with an assembly average burnup in excess of 30,000 MWD/MTU, the minimum best estimate margin available, relative to more limiting peaking values present in other assemblies, exceeds the corresponding rod bow penalties based on Ref. 14. Therefore, the staff concurs that sufficient available margin exists to offset rod bow penalties for assemblies with burnup greater than 30,000 MWD/MTU.

5.0 TECHNICAL SPECIFICATION CHANGES - Fuels, Physics, and Thermal-Hydraulics

The staff has reviewed the proposed modifications to the Technical Specifications for Cycle 2 operation as presented in Reference 1. The staff evaluation follows:

1. Specification 2.1.1.2 - The peak linear heat to centerline melt limit has been changed from 21.0 kW/ft to 22.0 kW/ft. This change is acceptable as discussed in Section 2.2 of this Safety Evaluation (SE).

2. Figure 2.1-1 - The thermal limit lines have been revised. This change reflects the approved reanalysis at 2700 MWT, the approved Technical Specification radial peaking factors and the implementation of approved margin recovery programs and is, therefore, acceptable.

3. Table 2.2-1 - The design reactor coolant flow has been changed from 370,000 gpm to 363,000 gpm. This is acceptable since all analyses that are sensitive to minimum flow requirements have been reanalyzed using the lower flow rate and have been reviewed and approved.

4. Figure 2.2-3 - The TM/LP LSSS has been revised. This change reflects the approved analysis at 2700 MWt, the approved Technical Specification radial peaking factors and the implementation of approved margin recovery programs and is, therefore, acceptable.

5. Figure 2.2-4 - This change is acceptable for the same reasons stated in 4. above.

6. Specifications B2.1.1, B2.2.1, and B3/4.2.5 - The value of minimum DNBR has been changed from 1.20 to 1.28. The new DNB limit has been derived using the Statistical Combination of Uncertainties (SCU) methodology (Ref. 3) which has been reviewed and approved in Section 7.0 of this SE and is therefore, acceptable. The initial request by FP&L to replace the actual minimum DNBR value by the phrase "the acceptable minimum DNBR limit" has been refused. The staff requires the bases to include both the value of 1.28 as well as reference to the use of SCU in its derivation.

7. Figure B2.1-1 - The axial power distributions used for thermal margin safety limits have been revised. This is acceptable since it reflects the approved higher radial peaking for Cycle 2 and the distributions have been derived using approved methods.

8. Specifications 3.1.1.2, 3.1.2.2, 3.1.2.4, 3.1.2.6, 3.1.2.8, B3/4.1.1.1, B3/4.1.1.2, B3/4.1.2 - The shutdown margin below 200°F has been changed from 2.0% k/k to 3.0% k/k. This is acceptable since it is consistent with the assumptions used in the approved safety reanalyses for those events that are affected by the change in shutdown margin.

9. Specification 3.1.3.1 - The number of CEA regulating groups has been changed from 6 to 5. This is acceptable for the reasons discussed in Section 3.3 of this SE.

10. Specification 3.1.3.1 - The time constraints on misaligned CEA have been revised to reflect a newly inserted figure (Fig. 3.1-1a) showing allowable time to realign a CEA vs. measured initial F_r^T. This is acceptable for the reasons stated in Section 6.4.2 in this SE concerning the CEA drop event.

11. Specification 3.1.3.4 - The CEA drop time from a fully withdrawn position to its 90% insertion position has been changed from 3.0 seconds to 2.7 seconds. This change is acceptable since it is consistent with plant measurements that have shown that the actual CEA drop time associated with a reactor trip is faster than previously assumed in the reference cycle.

12. Figure 3.1-2 - The CEA power dependent insertion limits (PDIL) have been revised. This is acceptable since it is consistent with the new CEA grouping changes discussed in Section 3.3 of this SE.

13. Figure 3.2-2 - The LHR excore LCO has been revised. This change reflects the approved reanalysis at 2700 MWt, the approved Technical Specification radial peaking factors and the implementation of approved margin recovery programs and is, therefore, acceptable.

14. Figure 3.2-3 - The allowable combinations of thermal power and F_r^T, F_{xy}^T have been revised. This revision reflects the higher peaking factors and power level used in the approved safety analyses and is, therefore, acceptable.

15. Specification 3.2.2 - The total planar radial peaking factor, F_{xy}^T, has been increased to 1.75 from 1.60. This is acceptable since it is appropriately accounted for in the nuclear design and the safety analyses and has been derived using approved methods.

16. Specification 3.2.3 - The total integrated radial peaking factor, F_r, has been increased to 1.70 from 1.60. This is acceptable since it is appropriately accounted for in the nuclear design and the safety analyses and has been derived using approved methods.

17. Specification 4.2.3.2, B3/4.2.2, B3/4.2.3, B3/4.2.4, and Table B3/4.2-1 - All references to rod bow penalty have been deleted. This is acceptable since the approved SCU methodology incorporates adjustments for rod bow directly in the DNBR limit rather than accounting for it explicitly in the monitoring of the radial peaking factor.

18. Figure 3.2-4 - The DNB LCO has been revised. This change reflects the approved reanalysis at 2700 MWt, the reactor coolant flow reduction to 363,000 gpm, the approved Technical Specification radial peaking factors, and the implementation of margin recovery programs and is, therefore, acceptable.

19. Table 3.2-2 - The upper bound of the cold leg temperature is increased from 548°F to 549°F and the reactor coolant flow rate is decreased from 370,000 gpm to 363,000 gpm. This is acceptable since calculations were performed to evaluate the impact of the changes on AOOs and postulated accidents and the results were found to be acceptable.

20. Table 3.3-5 - The feedwater isolation response time (total delay time) has been changed from 5.35 sec to 5.15 sec for both Containment Pressure - High and Steam Generator Pressure - Low initiating signals. This is acceptable since the surveillance requirements of specification 4.7.1.6 require verification of the 5.15 sec closure time periodically and this value has been used in the safety analyses for those transients affected by valve closing time.

21. Specification 3.4.3 - The maximum pressurizer indicated water level has been increased from 65% to 68%. This change has been accounted for in the approved analysis of a CVCS malfunction, which is the limiting event affected by this change. The change is, therefore, acceptable.

22. Specifications 3/4.7.1, B3/4.7.1.1, Table 3.7-1, and Table 3.7-2 - The pages have been revised. The changes made to maximum allowable power values reflect the revised analyses at 2700 MWt. The format of the specification has been changed to improve clarity. Therefore, these changes are acceptable.

23. Specification 3.7.1.6 - The full closure times of 5.6 sec and 5.35 sec for the main feedwater line isolation valves have been changed to 5.15 sec. These changes are acceptable since they have been assumed in the safety reanalyses. The peak containment pressure analysis used 5.15 sec as the closure time and gave acceptable results.

24. Specification B3/4.1.3 - The wording indicating at what power levels a DNBR SAFDL violation could occur has been removed and the wording on how this potential violation is eliminated has been clarified. Since the power levels at which a DNBR SAFDL violation may occur could vary slightly from cycle to cycle, this wording removal is acceptable.

25. Specification B3/4.1.3 - The steady state radial peak has been changed from 1.60 to 1.70. This is acceptable for the reasons stated in item 16 above.

26. Specification B3/4.1.3 - The reference to the actual radial peak for additional margin has been changed from $F_r^T = 1.50$ to F_r^T 1.70. Although there is a margin loss for the DNB-LSSS and the DNB-LCO due to the increased radial peaking, this is more than offset by margin gains due to the SCU, less severe axial power distributions for Cycle 2, use of a statistically based thermal hydraulic model, and a reduced required overpower margin (ROPM) for the limiting CEA subgroup drop event. The change is, therefore, acceptable.

27. Specification B3/4.1.3 - The allowable CEA misalignment time has been changed from 30 minutes for an F_r^T 1.50 to 60 minutes an initial F_r^T 1.55. This change is acceptable as it reflects the assumptions used in the reanalysis of the single CEA drop event.

28. Specification 5.3.1 - The reference to each fuel assembly containing 236 fuel rods has been changed to 236 fuel and poison rods. This is acceptable since Cycle 2 will contain assemblies with poison rods.

29. Specification 5.3.1 - The reference to a maximum total weight of 1698.5 grams uranium per fuel rod has been changed to approximately 1700 grams uranium. This is acceptable since variations in loading weights from cycle to cycle may occur and can be tolerated.

30. Specification 5.3.2 - The number of full-length CEAs contained in the core has been increased from 83 to 91. This is acceptable as it represents the addition of 8 full-length CEAs into vacant part-length CEA locations as discussed in Section 3.3 of this SE.

31. Specification 5.2.1 - The containment net free volume has been changed to 2.506×10^6 ft^3 from 2.5×10^6 ft^3. This is acceptable since it is based on a more detailed analysis of the containment net free volume.

6.0 SAFETY ANALYSIS

The design bases events (DBEs) considered in the safety analyses are categorized into two groups: anticipated operational occurrences (AOOs) and postulated accidents. All events were reanalyzed or re-evaluated for Cycle 2 to assure that the applicable criteria are met.

The AOOs are analyzed to assure that Specified Acceptable Fuel Design Limits (SAFDLs) on Departure from Nucleate Boiling (DNB) and Fuel Centerline to Melt (CTM) limits are not exceeded. These AOOs are divided into two categories. The first set requires Reactor Protection System (RPS) trips to assure that SAFDLs are not exceeded. The second set requires RPS trips and/or sufficient initial steady state margin (preserved by the LCOs) to prevent exceeding the SAFDLs. Transient analyses of the events in this latter category were performed utilizing the Statistical Combination of Uncertainties (SCU) methodology discussed in Section 7.0 of this SE.

Plant response to the DBEs was simulated using the same methods and computer programs as used and approved for Cycle 1 analyses or approved by the staff after Cycle 1 analyses. These include the CESEC III and STRIKIN II computer codes. Most events were reanalyzed to determine the effect of changes to key parameters from Cycle 1 to Cycle 2 such as an increase in rated core power, increases in radial power peaks and a lower minimum allowable reactor coolant flow.

6.1 Increase in Heat Removal Events

The licensee has evaluated the following AOOs that result in an increase in heat removal by the secondary system:

(a) decrease in feedwater temperature
(b) increase in feedwater flow
(c) increase main steam flow
(d) inadvertent opening of a steam generator safety valve or atmospheric dump valve.

The staff has reviewed the calculational models and assumptions used in the analyses of these events and find them acceptable. For all events, the maximum pressure within the reactor coolant system did not exceed 110% of the design pressure. Also, the minimum DNBR did not decrease below the design limit of 1.28 and the maximum local linear heat generation rate remained below the design limit of 22 kW/ft. The inadvertent opening of a steam generator safety valve is the limiting AOO that is analyzed for impact on offsite dose. The licensee has demonstrated conformance with the staff's acceptance criteria in the Standard Review Plan (SRP) Section 15.1.1, 15.1.2, 15.1.3, and 15.1.4. The staff, therefore, concludes that Cycle 2 operation is acceptable with respect to AOOs resulting in an increase in heat removal by the secondary system.

6.2 Decrease in Heat Removal Events

The licensee has evaluated the following AOOs that result in a decrease in heat removal by the secondary system:

(a) loss of external load
(b) turbine trip
(c) loss of condenser vacuum
(d) loss of normal AC power
(e) loss of normal feedwater

The staff has reviewed the calculational models and assumptions used in the analyses of these events and find them acceptable. The licensee has demonstrated that the limiting AOO that affects RCS pressure is the loss of condenser vacuum event. The peak RCS pressure attained is below the upset pressure limit of 110% of design pressure (2750 psia). The licensee has also shown that for the other AOOs leading to a decrease in heat removal by the secondary system, no fuel failure will occur, core geometry and CEA insertability are maintained with no loss of cooling capability, and maximum RCS pressure remains below 110% of design. The staff finds the results of these analyses in conformance with the acceptance criteria of SRP Sections 15.2.1 thru 15.2.7 and, therefore, acceptable.

6.3 Decrease in Reactor Coolant Flow Events

The licensee has analyzed both the partial and total loss of forced reactor coolant flow. The partial loss of forced reactor coolant flow is bounded by the total loss of forced reactor coolant flow and, therefore, only a detailed analysis of the latter was performed. This is the limiting AOO with respect to fuel integrity and is used to establish the minimum initial margin that must be maintained by the LCOs with respect to the DNBR limit. Therefore, this event results in an acceptable minimum DNBR of 1.28. The staff finds the plant response to a decrease in reactor coolant flow to be acceptable during Cycle 2 operation and in conformance with the staffs acceptance criteria of SRP Section 15.3.1.

6.4 Reactivity and Power Distribution Anomalies

6.4.1 Uncontrolled CEA Withdrawal Event

The licensee has analyzed the uncontrolled CEA withdrawal event from both high power and low power core conditions. The staff has reviewed the calculational models and the assumptions used in these analyses and find them acceptable. The licensee has shown that DNBR and fuel centerline melt SAFDLs are not violated and the RCS pressure remains below the upset limit. The staff, therefore, finds the results of an uncontrolled CEA withdrawal event during Cycle 2 to be in conformance with the acceptance criteria of SRP Section 15.4.1 and 15.4.2 and acceptable.

6.4.2 CEA Drop Event

The licensee has reanalyzed both the single and subgroup CEA drop event to determine the initial thermal margins that must be maintained by the LCOs such that the DNBR and CTM design limits will not be exceeded. The subgroup CEA drop was found to be more limiting. CEA withdrawal during the event is prohibited by the protection system so that power overshoot is not a problem.

The maximum initial radial peaking factor (F_r^T) assumed was the Technical Specification limit of 1.70. For the CEA subgroup drop, the maximum increase in F_r^T assumed was 19.0%. The comparable increase for a single CEA drop event is 14.0%. Therefore, the F_r^T can increase an additional 5% due to power redistribution following a single dropped CEA and still be bounded by the results of a subgroup CEA drop. The results of the licensee's analysis show that the net increase in F_r^T for the single drop after 15 minutes (18%) remains below the limiting increase in F_r^T for the subgroup drop (19%). After 63 minutes, the net increase in F_r^T is less than 19% above 1.70 when the pre-drop F_r^T is less than or equal to 1.54.

The licensee has shown that this event initiated from the Technical Specification LCOs will not exceed the DNBR and CTM design limits. The staff, therefore, finds the results to be in conformance with the acceptance criteria of SRP Section 15.4.3 and acceptable.

6.4.3 CVCS Malfunction (Inadvertent Boron Dilution)

The licensee has analyzed the boron dilution event to determine the setpoints of the startup channel alarms required for protection against loss of shutdown margin before the operator has time to stop the event. The event was analyzed from hot standby, hot shutdown, cold shutdown, and refueling conditions. The results indicate that the time available to the operator to stop the event from the alarm annunciation until criticality occurs meets the acceptance criteria stated in SRP Section 15.4.6 for minimum time from alarm annunciation to loss of shutdown margin. Therefore, the staff finds that St. Lucie 2 provides sufficient protection against inadvertent boron dilution events occurring during Cycle 2.

6.5 Increase in Reactor Coolant System Inventory

The licensee has identified the limiting increase in RCS inventory event to be the pressurizer level control system (PLCS) malfunction with a simultaneous closure of the letdown control valve to the zero flow position. This event is more limiting than the inadvertent operation of the emergency core cooling system (ECCS) because the shutoff head of the injection pumps is less than the RCS pressure during power operation. The operator has 20 minutes available after the high pressurizer level alarm occurs to prevent filling of the pressurizer. The staff finds this an acceptable period for operator action. Since operator action prevents a reactor and turbine trip, there is no event-related offsite dose and the peak RCS pressure is below 2415 psia. The increasing RCS pressure results in an increasing DNB and the fuel performance criterion is not approached. Therefore, the results of the analysis meet the acceptance criteria of SRP 15.5.1 and are acceptable.

6.6 Decrease in Reactor Coolant System Inventory

The inadvertent opening of a power operated relief valve (PORV) initiated at power was analyzed to demonstrate that this event does not result in violation of the SAFDLs and to determine a bias factor used in establishing the TM/LP trip setpoints. The event was also analyzed assuming a concurrent loss of offsite power. The minimum transient DNBR was 1.32 which is greater than the DNBR SAFDL limit of 1.28, thus no fuel failure is predicted. The plant is maintained in a stable condition due to automatic actions and, after 30 minutes, the operator opens the atmospheric dump valves and cools the plant to the point where shutdown cooling can be initiated. The staff finds the assumptions used and the analyses performed for this event to be acceptable and that the scenarios, as described by the licensee, assure that the most severe inadvertent opening of a PORV event has been considered.

6.7 Asymmetric Steam Generator Events

The four events that affect a single steam generator are:

(a) loss of load to one steam generator (LL/1SG)
(b) excess load to one steam generator (EL/1SG)
(c) loss of feedwater to one steam generator (LF/1SG)
(d) excess feedwater to one steam generator (EF/1SG)

Of these, the LL/1SG event is the limiting asymmetric event. This event is initiated by the inadvertent closure of a single main steam isolation valve (MSIV), which results in a loss of load to the affected steam generator. The asymmetric steam generator pressure trip (ASGPT) serves as the primary means of mitigating this transient with the steam generator level trip providing additional protection. The minimum transient DNBR calculated is greater than the DNBR SAFDL limit of 1.28. A maximum allowable LHGR of 18.1 kW/ft could exist as an initial condition without exceeding the fuel centerline melt SAFDL of 22.0 kW/ft during the transient. This amount of margin is assured by setting the LHR LCO based on the more limiting allowable LHR for LOCA of 13.0 kW/ft. The staff concludes that the calculations contain sufficient conservatism to assure that fuel damage will not result from any asymmetric steam generator event during Cycle 2 operation.

6.8 Conclusions

The licensee has presented results for various AOOs (with and without assumed single failures). The staff has reviewed the reanalyses and finds that they meet NRC acceptance criteria with respect to fuel and primary system performance. Therefore, adequate protection is provided for AOOs during Cycle 2 and the requirements of GDC 10, 15, and 26 are met.

6.9 Limiting Accidents

The licensee has reanalyzed events that, though not expected to occur during the lifetime of the plant, could have serious radiological consequences if not effectively mitigated. For accident conditions, the reactor coolant pressure should stay below the applicable ASME code limits. The core geometry should be maintained so that there is no loss of core cooling capability and control rod insertability. Radiological consequences must be well within the 10 CFR Part 100 limits.

6.10 Steam Systems Piping Failures Inside and Outside of Containment

Steam line breaks (SLB) inside containment may have break areas up to the cross section of the largest main steam pipe (6.305 ft^2). The licensee performed a parametric analysis in both MTC and break area and the limiting inside containment SLB event was found to be the break causing an effective flow area of 2.01 ft^2 with an effective MTC of $-.54 \times 10^{-4}$ /°F. A loss of AC power was postulated to accompany the SLB event. The results indicate that the number of fuel pins predicted to fail is less than 10% and thus a coolable geometry is maintained.

Break areas for outside containment SLBs are limited to the area of the flow restrictors (2.27 ft^2) located upstream of the containment penetrations. A parametric analysis in both MTC and break area identified the limiting event as the one which resulted in an flow area of 2.27 ft^2 with an effective MTC of -1.08×10^{-4} /°F. A loss of AC power was assumed to occur during the event. The results indicate that less than 10% of the fuel pins fail and consequently a coolable geometry is maintained. This is the most limiting postulated accident with respect to offsite dose and also with respect to fuel integrity.

The licensee has also performed analyses of the steamline break event to determine the potential for a post-trip return to power. The results of the steam line break event from HFP and HZP conditions with loss of offsite power show that there is no significant return to power.

The staff concludes that the consequences of postulated steam line break events meet the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage is limited such that CEA insertability would be maintained and that no loss of core cooling capability results. The requirements of GDC 31 and 35 demonstrating the integrity of the primary system and the adequacy of the ECCS have also been met. The parameters used as input were reviewed and found to be conservative and the model used has been previously reviewed and found acceptable by the staff. The staff, therefore, concludes that the licensee has demonstrated conformance with the acceptance criteria stipulated in SRP Section 15.1.5. As such, the staff concludes that the Cycle 2 operation is acceptable with respect to accidents resulting in breaks in the steam line.

6.11 Feedwater Line Break Event

The feedwater line break event with a loss of AC at time of high pressurizer pressure trip was analyzed. In order to maximize the radioactivity release during the transient, the analysis assumed that all of the initial activity in both steam generators and the activity added due to the primary to secondary leak rate tube leakage allowed by the Technical Specifications are released to the atmosphere with a decontamination factor of 1.0. The results show that the feedwater line break event with a loss of AC will not lead to a DNBR that is less than the design limit of 1.28 during the transient and the RCS peak presssure does not exceed 110% of design pressure. The staff, therefore, concludes that the results of a feedwater line break occurring during Cycle 2 meet the criterion of SRP Section 15.2.8 and are acceptable.

6.12 Single Reactor Coolant Pump Shaft Seizure/Sheared Shaft

The seized rotor event with loss of offsite power, Technical Specification steam generator tube leakage, failure to restore offsite power in 2 hours, and one stuck open atmospheric dump valve was analyzed. The results show that the number of fuel pins predicted to experience DNB is much less than 10%. Since only a small fraction of fuel pins fail, the staff finds that the results of a seized rotor event during Cycle 2 are acceptable and conform to the criteria of SRP Sections 15.3.3 and 15.3.4.

6.13 CEA Ejection Event

The range of initial conditions for a CEA ejection event examined by the licensee included zero power and full power with reactivity coefficients representative of BOC or EOC for these power level extremes. The analytical method employed in the reanalysis of this event is the NRC approved CE CEA ejection method. The results indicate that the maximum total energy deposited during the event is less than 280 cal/gm and, therefore, prompt fuel rupture with consequent rapid heat transfer to the coolant will not occur.

Although the licensee predicts no clad damage to occur, their criterion is an average enthalpy no greater than 200 cal/gm. The staff has continued using DNB as the criterion for clad failure. The staff has previously recommended the use of an assumed 10% amount of failed fuel in a radiological dose calculation for rod ejection transients in which DNB was not used as the clad failure mechanism and, therefore, continues to do so in this case. The predicted consequences of this event show that primary system integrity will be maintained and are, therefore, acceptable.

6.14 Loss of Coolant Accident (LOCA)

The ECCS performance evaluation for both the large break and the small break LOCA must show conformance with the acceptance criteria required by 10 CFR 50.46. The calculations were made using approved computer programs and models that meet the requirements of Appendix K to 10 CFR Part 50. The initial conditions were chosen to maximize the cladding temperature and oxidation. Containment parameters were chosen to minimize the calculated containment pressure to assure that the reflood calculations are conservatively calculated. The analyses account for an assumed amount of steam generator tube plugging of up to 300 average length tubes per steam generator.

For the large break analysis, the licensee analyzed both guillotine and slot breaks over a range of break sizes from 5.89 ft^2 to twice the flow area of the cold leg. The worst single failure is the loss of one of the low pressure safety injection (LPSI) pumps. From this analysis, the allowable peak linear heat generation rate (PLHGR) was determined to be 13.0 kW/ft with the 1.0 double ended guillotine break in the pump discharge leg identified as the limiting break. The results for Cycle 2 show a peak clad temperature of 2041°F, a peak local clad oxidation percentage of less than 13.3% and a peak core wide clad oxidation percentage of less than 0.55%. Since this meets the acceptance criteria for peak clad temperature, peak local clad oxidation percentage, and core wide clad oxidation percentage of 2200°F, 17.0%, and 1.0%, respectively, the staff concludes that operation of St. Lucie 2 with a PLHGR of 13.0 kW/ft provides acceptable results for the most limiting large break LOCA.

For the small break analysis, the licensee analyzed a spectrum of cold leg breaks in the reactor coolant pump discharge leg (0.5 ft^2, 0.1 ft^2, 0.0375 ft^2, and 0.015 ft^2). The worst single failure is the failure of one of the emergency diesel generators to start. Offsite power is assumed to be lost upon reactor trip. For an allowable PLHGR of 15.0 kW/ft, the 0.0375 ft^2 break was determined to be the limiting small break. The results show a peak clad temperature of 1740°F and a peak local clad oxidation percentage of less than 2%, which meet the acceptance criteria. The staff, therefore, concludes that operation of St. Lucie Unit 2 with a PLHGR of 15.0 kW/ft provides acceptable results for the most limiting small break LOCA.

Based on these results, the staff concludes that the LOCA analyses resulting from a spectrum of postulated piping breaks within the primary coolant pressure boundary are acceptable and meet the relevant requirements of 10 CFR 50.46 and Appendix K to 10 CFR Part 50. A comparison of the two limiting LOCA events demonstrates that the small break LOCA ECCS performance is less limiting than that for the large break LOCA performance results. Therefore, the staff concludes that operation of St. Lucie 2 with a PLHGR of 13.0 kW/ft is acceptable for Cycle 2.

7.0 STATISTICAL COMBINATION OF UNCERTAINTIES (SCU) METHODOLOGY

The procedures in the Statistical Combination of Uncertainties (SCU) methodology reviewed and approved by the NRC for St. Lucie 1 (Refs. 11, 12, and 13) have been applied by the licensee to St. Lucie 2 for Cycle 2 operation. Therefore, the review of the St. Lucie 2 SCU was directed mainly toward the plant-specific application that accommodates the differences in plant design and reactor protection systems. The methodology consists of three parts. Part 1 (Ref. 11) describes the application of the SCU methods to the development of the local power density (LPD) and TM/LP limiting safety system settings (LSSS). These are used in the analog reactor protection system to protect against fuel centerline melt and DNB. Part 2 (Ref. 12) combines the uncertainties associated with the reactor system parameters to develop a revised DNBR limit corresponding to the SAFDL to be used in the plant safety analysis and the evaluation of the LSSS and the LCOs. Part 3 (Ref. 13) uses the SCU methodology to calculate LHR and DNB LCOs.

The plant independent calculational-measurement uncertainties used in the St. Lucie 2 SCU were derived from recent data from Cycle 5 of St. Lucie 1, Cycles 5 and 6 of Calvert Cliffs 1, and Cycles 4 and 5 of Calvert Cliffs 2 which has been obtained after the SCU reports were issued. The plant specific St. Lucie 2 data for the instrument circuitry, the lead bank CEA configuration and power dependent insertion limit were used to evaluate the plant dependent uncertainties. The shape annealing factor (SAF) component of the shape index uncertainty developed for St. Lucie 1 was used for St. Lucie 2. The licensee will evaluate the need to measure the St. Lucie 2 SAFs prior to Cycle 2 startup.

The licensee has provided the St. Lucie 2 component uncertainties associated with the LPD LHR and DNB LCOs and the LHR and TM/LP LSSSs. This data is analogous to that which had been provided previously for St. Lucie 2 and approved by the NRC.

Since the uncertainty values used in this analysis have been justified with the appropriate sources and the combination of these uncertainties is performed with the approved methods, the staff concludes that the overall aggregate uncertainty factors derived for the TM/LP and LPD LSSS are acceptable.

The statistically derived MDNBR limit contains various allowances, or penalties, as described in Ref. 12. In addition to these, an additional 5% penalty on the CHF standard deviation due to the effect of prediction uncertainty in the CHF correlation in the calculation of the DNBR limit as well as a 5% code uncertainty were included. These have been required by the NRC in previous SCU reviews. After including a 1.75% MDNBR rod bow penalty plus a 0.01 DNBR penalty due to the HID-1 grid design, the MDNBR was determined to be 1.279.

The staff finds that the plant specific parameters of St. Lucie 2 have been properly applied with the SCU methodology previously reviewed and approved by the NRC and that appropriate adjustments in the form of penalties have been included. The proposed DNBR value of 1.28 still provides at least a 95% probability at a 95% confidence level that DNB does not occur on a fuel rod having that minimum DNBR. Therefore, the staff concludes that the minimum DNBR limit of 1.28 is acceptable for the St. Lucie 2 Cycle 2 reload application.

Cycle 2 operation within the DNB and LHR LCOs must provide the necessary initial DNB and LHR margins to prevent exceeding the acceptable limits during DBEs where changes in DNBR and LHR are important. The methods for statistically combining the uncertainties involved in these LCOs are similar to those used for determining the LSSS limits. In order to determine the LCO required overpower margin (ROPM), the loss of coolant flow (LOF) and full length CEA drop events were analyzed for St. Lucie 2. The licensee has determined that these two events are bounding for the DBEs requiring intervention of RPS trips and/or sufficient initial steady state thermal margin to prevent exceeding the acceptable limits. The analyses for these limiting ROPM events discussed in the safety analysis section of this SE (Sections 6.3 and 6.4.2) were initiated from nominal conditions. The ROPM calculated at nominal conditions is then combined with the incremental ROPM, defined by these SCU transient analyses to determine the final ROPM, which must be incorporated into the protection and monitoring system setpoints.

The cycle independent maximum incremental ROPM deviations determined by these SCU transient analyses were developed using the methodology previously reviewed and approved in Ref. 13. The results appear to be consistent with the results reported therein. Therefore, the staff concludes that the statistically combined uncertainties described for St. Lucie 2 are acceptable for the DNB and LHR LCO calculations.

The application of the SCU methods described is acceptable for the St. Lucie 2 reload calculations. The overall aggregate uncertainties presented for the TM/LP LSSS and LPD LSSS are acceptable for the St. Lucie 2 trip setpoint calculations. The SCU equivalent minimum DNBR limit of 1.28 is acceptable for the reload analyses. The statistically combined uncertainties presented for the DNB and LHR LCO calculations are acceptable. However, if future reloads use computer codes and correlations other than those described in this application, a reanalysis of the aggregate uncertainties for the LSSS and LCO and the minimum DNBR limit will be required.

8.0 EVALUATION FINDINGS - Fuels, Physics, and Thermal-Hydraulics

The staff has reviewed the fuels, physics and thermal-hydraulics information presented in the St. Lucie 2 Cycle 2 reload report, the Technical Specification revisions, and the safety reanalyses and the uncertainties derived for St. Lucie 2 Cycle 2 by the SCU methodology. Based on the evaluations given in the preceding sections, the staff finds the proposed reload and associated modified Technical Specifications acceptable.

There is a license condition resulting from the staff review of fuel rod axial growth that is discussed in Section 2.1 of this SE. A formal report addressing this will be submitted by the licensee to the NRC prior to Cycle 2 startup.

9.0 CONTAINMENT

9.1 Containment Evaluation and Findings

In the licensee's report, the impact of a proposed power upgrade from 2560 to 2700 MWt on the various containment related analyses was presented. The affected analyses include the containment pressure and temperature response for the design basis LOCA and MSLB, subcompartment pressurization, ECCS back pressure calculation, and hydrogen generation. As a result of the containment analysis, several changes to the plant Technical Specifications are necessary to accommodate the proposed power increase. These changes are addressed in Section 9.2, herein.

The licensee has performed containment analyses similar to those presented in the FSAR for Cycle 1 operation. The most limiting LOCA and MSLB cases identified in the FSAR, were reanalyzed by the licensee. In so doing, the mass and energy release data were changed to reflect the increase in power level; the containment spray actuation setpoint, start time and flow rate were adjusted to compensate for the revised blowdown data. The staff has reviewed the initial conditions and assumptions used for peak containment pressure and temperature calculations and finds them acceptable. The calculated peak containment pressure and temperature for the MSLB accident are 43.7 psig and 413.9°F, respectively, and for the LOCA are 42.7 psig and 265.8°F, respectively. These values are below the design conditions of 44 psig and 420°F.

The licensee has also evaluated the impact of the power upgrade on subcompartment loading. Based on the large margin to design (> 100%) of the compartment loading, shown in the FSAR, and the small increase (< 0.5%) in the peak pressure in the containment reanalysis, the licensee concludes that the subcompartment loading would remain below design values; the staff concurs with the licensee's conclusion.

The impact of the power increase on the post-LOCA hydrogen build-up inside containment has been re-analyzed. Results of the analysis show that a single recombiner started 50 hours after the accident is sufficient to limit the hydrogen concentration in containment to below the Regulatory Guide 1.7 lower flammability limit of 4.0 volume percent. The administrative procedures described in the FSAR Section 6.2.5.2.2 require the operator to start the recombiner within 24 hours following a LOCA. In addition, the operator is alerted by alarms from the containment hydrogen analyzer system at 3.0% hydrogen concentration, which should occur no sooner than about 50 hours after onset of the accident. Based on the foregoing discussion, the staff concurs with the licensee that the existing combustible gas control system is capable of preventing the hydrogen gas concentration inside containment from exceeding the lower flammability limit.

9.2 Technical Specification Changes - Containment

1. The containment spray high-high trip setpoint has been lowered from 9.30 psig to 5.40 psig, and the allowable value has been lowered from 9.40 psig 5.50 psig. Lowering of the containment spray setpoint will result in lower peak containment pressures following mass and energy releases to the containment under power increase conditions. For the containment reanalysis, a conservative trip setpoint value of 6.0 psig was used for containment spray actuation, and the calculated peak containment pressure was below the design value. The staff, therefore, finds the proposed change in the containment spray trip setpoint acceptable.

2. The allowable response time for containment pressure instrumentation has been reduced from 1.55 seconds to 1.15 seconds. This reduction in response time is based on in-plant experience with instrument performance; therefore, the staff finds this change acceptable.

3. The high containment pressure trip setpoint for actuation of Engineered Safety Features (ESF) functions has been lowered from 5.0 psig to 4.7 psig, with the allowable value being reduced from 5.1 psig to 4.8 psig. The ESF functions affected include safety injection, containment isolation, and main steam line isolation. With regard to containment isolation, Item II.E.4.2 of NUREG-0737 recommends that the containment setpoint pressure for initiating the isolation of non-essential lines penetrating containment be reduced to the minimum value compatible with normal operating conditions. Based on a telecon with the licensee on October 25, 1984, a setpoint of 3.5 psig, instead of 4.7 psig, was proposed by the licensee. This change will comply with the requirements of Item II.E.4.2 of NUREG-0737 and is acceptable to the staff. The licensee has agreed to formally document the justification for the proposed setpoint value.

4. The feedwater isolation signal response time has been lowered from 5.35 seconds to 5.15 seconds. This change reflects the closure time for the main feedwater isolation valves based on operating experience. A valve closure time of 5.15 seconds was assumed in the peak containment pressure analysis; therefore, the staff finds this change acceptable.

10.0 ENVIRONMENTAL CONSIDERATION

This amendment involve changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

11.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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12.0 REFERENCES

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