DMB-016

December 9, 1983

:ket Nos. 50-250
and 50-251

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Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408

Dear Mr. Uhrig:

The Commission has issued the enclosed Amendment No. 98 to Facility Operating License No. DPR-31 and Amendment No. 92 to Facility Operating License No. DPR-41 for the Turkey Point Plant Unit Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated June 3, 1983, supplemented on November 16, 1983.

These amendments involve Technical Specification changes to support planned fuel design modification during Cycle 9 refueling for Unit 3, Cycle 10 refueling for Unit 4 and subsequent cycles. It is planned to replace the Westinghouse 15 x 15 low-parasitic (LOPR) fueled cores with Westinghouse 15 x 15 optimized fuel assembly (OFA) core with Wet Annular Burnable Absorber (WABA) Rods. The Technical Specifications allow (1) increases in shutdown and control rod drop time which will be based on safety analysis for the transition cores; (2) use of burnable poison rods of an approved design for reactivity and/or power distribution factors; and (3) changes in hot channel factors and other power distribution factors affecting departure from nucleate boiling (DNB). The change in core physics parameters and thermal characteristics are required due to the improved neutronic characteristics of fuel assemblies and fuel management considerations.

The request for these amendments was noticed on July 20, 1983 (48 FR 33080) and no petition for leave to intervene or significant hazards consideration comments were received pursuant to that notice. However, a petition for leave to intervene and comments were received on a separate request for amendments, which were noticed on October 7, 1983 (48 FR 45862), relating to different aspects of the core reload design. Some of these comments and concerns were relevant to the present amendments. Since these amendments had not yet issued,

Dr. Robert E. Uhrig, Vice President
Advanced Systems and Technology
Florida Power and Light Company

the staff, in its discretion, has chosen to address the comments relevant to these amendments. The comments and concerns were received from the Center for Nuclear Responsibility and Ms. Joette Lorion.

Copies of the Safety Evaluation and Notice of Issuance and Final Determination of No Significant Hazards Consideration are enclosed.

Sincerely,

ORIGINAL SIGNED BY

Daniel G. McDonald, Jr., Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures
1. Amendment No. 98 to DPR-31
2. Amendment No. 92 to DPR-41
3. Safety Evaluation
4. Notice

cc w/enclosures: See next page

cp 12/8/83

w/noted revisions to letter and SE

ORB#1:DL	ORB#1:DL	C-ORB#1:DL	OELD	AD:OR:DL
CParrish	DMcDonald;ps	SVarga	MYoung	GLainas
12/6/83	12/10/83	12/10/83	12/7/83	12/9/83

Robert E. Uhrig
Florida Power and Light Company

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Units 3 and 4

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.98
License No. DPR-31

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated June 3, 1983, supplemented on November 16, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 98 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 9, 1983

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.92
License No. DPR-41

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Florida Power and Light Company (the licensee) dated June 3, 1983, supplemented on November 16, 1983, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-41 is hereby amended to read as follows:

   (B) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 92 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of startup of Cycle 10.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: December 9, 1983

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.2-2	3.2-2
B3.2-2	B3.2-2
5.2-1	5.2-1
B2.1-1	B2.1-1
B2.1-2	B2.1-2
B2.3-2	B2.3-2
B2.3-3	B2.3-3
B3.1-1	B3.1-1
B3.2-3	B3.2-3
B3.2-8	B3.2-8

f. Except for low power physics tests, the shutdown margin with allowance for a stuck control rod shall exceed the applicable value shown on Figure 3.2-2 under all steady-state operating conditions from zero to full power, including effects of axial power distribution. The shutdown margin as used here is defined as the amount by which the reactor core would be subcritical at hot shutdown conditions (540°F) if all control rods were tripped, assuming that the highest worth control rod remained fully withdrawn, and assuming no changes in xenon, boron concentration or part-length rod position.

g. During physics tests and control rod exercises, the insertion limits need not be met, but the required shutdown margin, Figure 3.2-2 must be maintained or exceeded.

2. MISALIGNED CONTROL ROD
If a part length* or full length control rod is more than 12 steps out of alignment with its bank, and is not corrected within 8 hours, power shall be reduced so as not to exceed 75% of interim power for 3 loop or 45% or interim power for two loop operation, unless the hot channel factors are shown to be no greater than allowed by Section 6a of Specification 3.2

3. <u>ROD DROP TIME</u>
The drop time of each control rod shall be no greater than 2.4 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

4. <u>INOPERABLE CONTROL RODS</u>
a. No more than one inoperable control rod shall be permitted during sustained power operation, except it shall not be permitted if the rod has a potential

* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

The various control rod banks are each to be moved as a bank, that is, with all rods in the bank within one step (5/8-inch) of the bank position. The control system is designed to permit individual rod movement for test purposes. Position indication is provided by two methods: a digital count of actuating pulses which shows the demand position of the banks and a linear position indicator (LVDT) which indicates the actual rod position.[(2)] The relative accuracy of the linear position indicator (LVDT) is such that, with the most adverse error, an alarm will be actuated if any two rods within a bank deviate by more than 15 inches. In the event that an LVDT is not in service, the effects of a malpositioned control rod are observable on nuclear and process information displayed in the control room and by core thermocouples and in-core movable detectors. Complete rod misalignment (part-length* or full-length control rod 12 feet out of alignment with its bank) does not result in exceeding core limits in steady-state operation at rated power. If the condition cannot be readily corrected, the specified reduction in power to 75% (3 loop) or 45% (2 loop) will insure that design margins to core limits will be maintained under both steady-state and anticipated transient conditions. The 8-hour permissible limit on rod misalignment is short with respect to the probability of an independent accident. The 24-hour period ensures that no significant burnup effects would be caused by the inserted rod.

The specified rod drop time is consistent with safety analyses that have been performed.[(X)]

The In-Core Instrumentation has five drives with detectors each of which has ten thimbles assigned[(3)]. This provides broad capability for detailed flux mapping.

The ion chambers located outside the reactor vessel measure flux distribution at the top and bottom of the core. Core traverses in a few of the in-core instrument paths will establish that the fixed flux measurement equipment is properly calibrated.

Operating experience has established that the flux measurement system is of a reliable design, and that the 10% load reduction, in the event of recalibration delay, is ultra conservative compensation.

References:
(1) FSAR - Section 14
(2) FSAR - Section 7.2
(3) FSAR - Section 7.6
(X) FPL licensing submittal for transition cores to the NRC

* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

5.2 REACTOR

REACTOR CORE

1. The reactor core contains approximately 71 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy - 4 tubing to form fuel rods. The reactor core is made up of 157 fuel assemblies. Each fuel assembly contains 204 fuel rods.

2. The average enrichment of the initial core is a nominal 2.50 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.10 weight per cent of U-235.

3. Reload fuel will be similar in design to the initial core. The enrichment of reload fuel will be no more than 3.5 weight per cent of U-235.

4. Burnable poison rods in the form of rod clusters, which are located in vacant rod cluster control guide tubes are used for reactivity and/or power distribution control.

5. There are 45 full-length RCC assemblies and 8 partial-length* RCC assemblies in the reactor core. The full-

* Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

B2.1 Bases for Safety Limit, Reactor Core

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and Reactor Coolant Temperature and Pressure have been related to DNB. This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability with 95% confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

The curves of Figures 2.1-1, 2.1-1a, and 2.1-1b show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the calculated DNBR is no less than the design DNBR value or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid.

The curves are based on a enthalpy hot channel factor, $F^N_{\Delta H}$, of 1.55 and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$$F^N_{\Delta H} \leq 1.55 \; [1 + 0.2 \; (1 - P)]$$

where P is the fraction of RATED THERMAL POWER.

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit assuming the axial power imbalance is within the limits of the $f(\Delta q)$ function of the Overtemperature ΔT trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core safety limits.

The $f(\Delta q)$ function in the Overpower ΔT and Overtemperature ΔT protection system setpoints includes effects of fuel densification on core safety limits. The setpoints will ensure that the safety limit of centerline fuel melt will not be reached and the applicable design limit DNBR will not be violated.[10]

Pressurizer

The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident.[6]

The high pressurizer pressure reactor trip is set below the set pressure of the pressurizer safety valves and limits the reactor operating pressure range. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified setpoint allows margin for instrument error[3] and transient level overshoot before the reactor trips.

Reactor Coolant Flow

The low flow reactor trip protects the core against DNB in the event of loss of one or more reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis.[7] The low frequency and under voltage reactor trips protect against a decrease in flow. The specified setpoints assure a reactor trip signal before the low flow trip point is reached. The underfrequency trip setpoint preserves the coastdown energy of the reactor coolant pumps, in case of a system frequency decrease, so DNB does not occur. The undervoltage trip setpoint will cause a trip before the peak motor torque falls below 100% of rated torque.

Steam Generators

The low-low steam generator water level reactor trip assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting of the auxiliary feedwater system.[8]

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

Reactor Trip Interlocks

Specified reactor trips are by passed at low power where they are not required for protection and would otherwise interfere with normal operation. The prescribed set points above which these trips are made functional assures their availability in the power range where needed.

An automatic reactor trip will occur if any pump is lost above 55% power which will prevent the minimum value of the DNBR from going below the applicable design limit during normal and anticipated transient operations when only two loops are in service,(9) and the overtemperature ΔT trip setpoint is adjusted to the value specified for three loop operation.

Reset of reactor trip interlocks will be done under strict administrative control.

References

(1) FSAR 14.1.1
(2) FSAR 14.1.2
(3) FSAR 14.1
(4) FSAR 7.2, 7.3
(5) FSAR 3.2.1
(6) FSAR 14.3.1
(7) FSAR 14 (page 14-30 and 14.1.9)
(8) FSAR 14.1.11
(9) FSAR 14.1.9
(10) WCAP-8074

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

B.3.1 BASES FOR LIMITING CONDITIONS FOR OPERATION, REACTOR COOLANT SYSTEM

1. Operational Components

The specification requires that significant number of reactor coolant pumps be operating to provide coastdown core cooling in the event that a loss of flow occurs. The flow provided will keep DNBR well above the applicable design limit. When the boron concentration of the Reactor Coolant System is to be reduced the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the reactor coolant system volume in approximately one half hour.

Each of the pressurizer safety valves is designed to relieve 283,300 lbs. per hr. of saturated steam at the valve setpoint Below 350°F and 450 psig in the Reactor Coolant System, the Residual Heat Removal System can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available the amount of steam which could be generated at safety valve lifting pressure would be less than the capacity of a single valve. Also, two safety valves have capacity greater than the maximum surge rate resulting from complete loss of load.[2]

The 50°F limit on maximum differential between steam generator secondary water temperature and reactor coolant temperature assures that the pressure transient caused by starting a reactor coolant pump when cold leg temperature is ≤ 275°F can be relieved by operation of one Power Operated Relief Valve (PORV). The 50°F limit includes instrument error.

The plant is designed to operate with all reactor coolant loops in operation, and maintain DNBR above the applicable design limit during all normal operations and anticipated transients. In power operation with one reactor coolant loop not in operation this specification requires that the plant be in at least Hot Shutdown within 1 hour.

In Hot Shutdown a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be operable.

In Cold Shutdown, a single reactor coolant loop or RHR coolant loop provides sufficient heat removal capability for removing decay heat, but single failure considerations require that at least two loops be operable. Thus, if the reactor coolant loops are not operable, this specification requires two RHR loops to be operable.

Design criteria have been chosen for normal and operating transient events which are consistent with the fuel integrity analyses. These relate to fission gas release, pellet temperature and cladding mechanical properties. Also, the minimum DNBR in the core must not be less than the applicable design limit in normal operation or in short term transients.

In addition to conditions imposed for normal and operating transient events, the peak linear power density must not exceed the limiting Kw/ft values which result from the large break loss of coolant accident analysis based on the ECCS Acceptance Criteria limit of 2200°F. This is required to meet the initial conditions assumed for loss of coolant accident. To aid in specifying the limits on power distribution, the following hot channel factors are defined.

$F_Q(Z)$, <u>Heat Flux Hot Channel Factor</u>, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

F_Q^E, <u>Engineering Heat Flux Hot Channel Factor</u>, is defined as the allowance on heat flux required for manufacturing tolerances. The engineering factor allows for local variations in enrichment, pellet density and diameter, surface area of fuel rod and eccentricity of the gap between pellet and clad. Combined statistically the net effect is a factor of 1.03 to be applied to fuel rod surface heat flux.

$F_{\Delta H}^N$, <u>Nuclear Enthalpy Rise Hot Channel Factor</u>, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to average rod power.

It should be noted that $F_{\Delta H}^N$ is based on an integral and is used as such in the DNB calculations. Local heat fluxes are obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F_{\Delta H}^N$.

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

$$W(Z) = \text{Max}\left(\frac{F_Q(Z)(\text{Base Load Case(s), 150 MWD/T})}{F_Q(Z)(\text{ARO, 150 MWD/T})}, \frac{F_Q(Z)(\text{Base Case(s), 85\% EOL BU})}{F_Q(Z)(\text{ARO, 85\% BOL BU})}\right)$$

For Radial Burndown operation the full spectrum of possible shapes consistent with control to a ± 5% ΔI band needs to be considered in determining power capability. Accordingly, to quantify the effect of the limiting transients which could occur during Radial Burndown operation, the function $F_z(Z)$ is calculated from the following relationship:

$$F_z(Z) = [F_Q(Z)]_{\text{FAC Analysis}}/[F_{xy}(Z)]_{\text{ARO}}$$

As discussed above, the essence of the procedure is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This can be accomplished without part length rods* by using the boron system to position the full length control rods to produce the required indicated flux difference.

For Operating Transient events, the core is protected from overpower and a minimum DNBR of less than the applicable design limit by an automatic protection system. Compliance with operating procedures is assumed as a precondition for Operating Transients; however, operator error and equipment malfunctions are separately assumed to lead to the cause of the transients considered.

Above the power level of P_T, additional flux shape monitoring is required. In order to assure that the total power peaking factor, F_Q, is maintained at or below the limiting value, the movable incore instrumentation will be utilized. Thimbles are selected initially during startup physics tests so that the measurements are representative of the peak core power density. By limiting the core average axial power distribution, the total power peaking factor F_Q can be limited since all other components remain relatively fixed. The remaining part of the total power peaking factor can be derived based on incore measurements, i.e., an effective radial peaking factor $\overline{R}$, can be determined as the ratio of the total peaking factor resulting from a full core flux map and the axial peaking factor in a selected thimble.

*Any reference to part-length rods no longer applies after the part-length rods are removed from the reactor.

References
FSAR - Section 14.3.2

This amendment effective as of date of issuance for Unit 3 and date of startup, Cycle 10, Unit 4.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 98 TO FACILITY OPERATING LICENSE NO. DPR-31
AND AMENDMENT NO.92 TO FACILITY OPERATING LICENSE NO. DPR-41
FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT PLANT UNIT NOS. 3 AND 4
DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By letters dated June 3, 1983, and supplemented on November 16, 1983, to provide additional information, Florida Power & Light Company submitted a request (Ref. 1) for an amendment of the Technical Specifications contained in Appendix A of Facility Operating Licenses DPR 31 and 41. The Technical Specification changes are intended to accommodate: (1) a planned fuel design change from the Westinghouse (W) 15X15 low parasitic (LOPAR) design to the 15X15 Optimized Fuel Assembly (OFA), and (2) use of Wet Annular Burnable Absorber (WABA) rods.

2.0 FUEL MECHANICAL DESIGN

Turkey Point Units 3 and 4 have been operating with all W 15X15 low parasitic (LOPAR) fuel. The Unit 3 Cycle 9 and Unit 4 Cycle 10 cores will include W 15X15 OFAs resulting in a 1/3 OFA-2/3 LOPAR mixture. Subsequent reloads are expected to eventually contain only OFA fuel. Although the W 15X15 OFA fuel is a new design, it is very similar to the W 15X15 standard low parasitic (LOPAR) fuel design. The major change introduced by the 15X15 OFA design is the use of 5 intermediate Zircaloy grids replacing 5 intermediate Inconel grids in the LOPAR fuel. The Zircaloy grids have thicker and wider straps than the Inconel grids in order to closely match the Inconel grid strength. Furthermore, the 15X15 OFA Zircaloy grid design is similar to the W 17X17 OFA grid design, which was described in Westinghouse Report No. WCAP-9500-A. This report has been reviewed and approved by the NRC staff (Ref. 2).

In performing our review of the 15X15 OFA fuel for Turkey Point Units 3 and 4, we have relied upon the D. C. Cook Unit 1 Cycle 8 reload report (Ref. 3) that the design criteria and evaluation methods used for 17X17 OFA in WCAP-9500-A were also used for 15X15 OFA. This information is also applicable to Turkey Point Units because identical fuel is used. The balance of our review thus focused on those plant-specific issues identified in the SER for WCAP-9500-A insofar as they are applicable to Turkey Point Units 3 and 4. Our evaluation of those issues follows.

2.1 CLADDING COLLAPSE

The licensee uses an approved method described in Westinghouse Report No. WCAP-8377 (Ref. 4) to analyze cladding collapse. The result for Turkey Point shows that no cladding collapse is expected up to 40,000 EFPH (about 51,200 MWd/MTU peak-rod average burnup) for the new W fuel design. We conclude, therefore, that no cladding collapse is expected for the proposed and subsequent cycles of operation.

2.2 FUEL THERMAL CONDITIONS

The Turkey Point submittal is based, in part, upon fuel thermal analyses generated with a revised (Ref. 5) version of a previously approved W code called PAD (Ref. 6). This revision has been approved for generic reference in W fuel design (including OFA) calculations.

2.3 CLADDING SWELLING AND RUPTURE

For large break loss-of-coolant accident analysis, the licensee used the approved 1981 large break Emergency Core Cooling System (ECCS) evaluation model (Ref. 7), which includes approved cladding swelling and rupture models. The use of this ECCS model obviates the need for supplemental ECCS calculations mentioned in the SER for WCAP 9500-A (Ref. 2). We thus find that cladding swelling and rupture have been adequately treated in the ECCS analysis.

2.4 SEISMIC AND LOCA LOADS

In 1975 asymmetric blowdown forces on PWRs during LOCA was identified. As a result, NRC Report No. NUREG-0609 (Asymmetric Blowdown Loads on PWR Primary Systems, Unresolved Safety Issue A-2) was issued to address this concern and required all PWRs to submit such an analysis for evaluating fuel assembly structural adequacy.

Westinghouse A-2 Owners Group, including Turkey Point Units 3 and 4, submitted two reports, WCAP-9558, Revision 2 and WCAP-9787 (Ref. 8), for staff review in response to NUREG-0609. They stated that a rapid blow-down is very unlikely because the stainless steel primary piping would leak before it breaks during a LOCA; therefore, the reports argue that the requirements of NUREG-0609 can be waived.

Although the review of W A-2 Owners Group reports has not yet been completed, no structural response analysis of combined seismic and LOCA loads is presently being required from any A-2 Owner.

Although the issue of combined seismic and LOCA loads need not be resolved at this time, the analysis requirement remains for the seismic event alone. This is particularly so because the new OFA fuel assemblies and the existing LOPAR fuel assemblies have slightly different structural properties as a result of incorporation of the new Zircaloy grid. Since OFA and LOPAR fuels will both be loaded in mixed configurations during the next few operating cycles, the licensee analyzed several mixed configurations for structural adequacy. Generic methods (WCAP-9401), which were previously reviewed and approved by NRC, were used for this analysis. Results show at least 20 percent margin relative to allowable limits for the spacer grid and more than a factor of 3 margin for other fuel assembly components including the functionally important thimble tubes. Based on the finding of adequate margins and the use of approved methods, we conclude that fuel assembly structural adequacy has been demonstrated.

2.5 WET ANNULAR BURNABLE ABSORBERS

Turkey Point Units 3 and 4 will utilize a new burnable poison design, the WABA rods, in the future cores. In fact, these rods contain no fuel and by adding poison rods to the outer rows of tubes, they modify flux characteristics such that the current Technical Specifications for the hot channel limit and total peaking factor will limit the total reactor power level to less than 100 percent. The licensee has a proposed amendment request which will permit operation at full power which is currently under review. The WABA rod design consists of annular pellets of aluminum oxide and boron carbide (Al_2O_3-B_4C) burnable absorber material encapsulated within two concentric Zircaloy tubings. The reactor coolant flows inside the inner tubing and outside the outer tubing of the annular rod. The topical report describing the WABA design (Ref. 9) has been recently reviewed and approved (Ref. 10), and the utilization of WABA rods in both units would thus be automatically approved subject to certain conditions described in the NRC staff's approval of the generic topical report (those conditions concern surveillance and the analysis of core bypass flow).

The WABA surveillance is discussed in Section 2.7 and the analysis of core bypass flow is discussed in Section 4.0 of this Safety Evaluation.

2.6 GUIDE THIMBLE DIAMETER REDUCTION

The 15X15 OFA guide thimbles are similar in design to those in the LOPAR fuel assemblies except for a 13 mil reduction in the inside diameter (ID) and outside diameter (OD) of the guide thimble above the dashpot. Because guide thimble tube fretting wear has been observed in some PWR designs, the NRC staff questioned the potential for increased wear in the OFA 15X15 design due to the reduction in clearance for the control rods.

However, W has shown in other cases (Ref. 11) that results of the analysis for OFA 15X15 guide thimble tube wear, using an approved technique (Ref. 12), were unchanged in the predicted guide tube wear compared to the W 15X15 standard design. Based on the information presented, we agree with the licensee that the reduction in guide thimble to rod control cluster (RCC) rodlet clearance should have no adverse effect on the extent of guide tube wear and, consequently, there is reasonable assurance that (a) the structural integrity of the 15X15 OFA will be maintained with respect to load carrying capability of the guide thimble tubes, and (b) "scramability" will be maintained.

2.7 POST-IRRADIATION SURVEILLANCE

As indicated in Standard Review Plan (SRP) Section 4.2.II.D.3, a post-irradiation fuel surveillance program should be established to detect anomalies or confirm expected fuel performance.

For a new fuel design, such as the 15X15 OFA, we normally request that a fuel surveillance program be developed for the first two lead plants utilizing the new design. Since two other operating reactors (other than Turkey Point Units 3 and 4) have been identified as lead plants for the 15X15 OFA/WABA design, we conclude that no special surveillance requirements are necessary for this fuel design change at Turkey Point.

As for the WABA rods, the licensee has committed to have a supplementary surveillance program as described in Reference 10 if Turkey Point is the first or second lead plant to discharge WABA rods. We find this acceptable.

2.8 SUMMARY

We have reviewed the fuel assembly mechanical design for Turkey Point Units 3 and 4. We conclude that the fuel mechanical design, which includes the W 15X15 OFAs and the WABAs, is acceptable.

3.0 NUCLEAR DESIGN

The proposed Technical Specification changes will allow the transition from low parasitic 15X15 (LOPAR) assemblies to 15X15 OFA assemblies. These OFA assemblies are identical to the LOPAR assemblies except that five of the interior Inconel grids have been replaced by Zircaloy grids. The physics characteristics for the OFA fuel are only slightly different from those of the LOPAR. These differences are within the normal range of variations seen from cycle to cycle. They are due primarily to fuel management considerations and not due to the fuel assembly design.

The 15X15 OFA has features similar to the W 17X17 OFA which has been generically approved by NRC (Ref. 2). There has been experience with the OFA fuel design configurations and recently the 15X15 OFA design configuration was approved for the D. C. Cook Unit 1 Cycle 8 core. The standard calculational methods as described in Reference 13 continue to apply. Each reload core will be evaluated to assure that design and safety limits are satisfied according to the reload methodology. On this basis we approve use of the 15X15 OFA design for Turkey Point Units 3 and 4. Acceptability of the WABA design is discussed in Section 2.5.

4.0 THERMAL HYDRAULIC EVALUATION

Since the Turkey Point Units 3 and 4 cores will be refueled with the 15X15 OFA fuel and the WABA rods, these cores will have LOPAR-OFA mixed core configurations during the transition fuel cycles. The 15X15 OFA fuel has design features similar to the 15X15 LOPAR fuel except for the use of 5 intermediate Zircaloy grids of the OFA fuel to replace the 5 intermediate Inconel grids used in the LOPAR fuel. The Zircaloy grids have thicker and wider grid straps which result in the OFA fuel assembly having approximately 4.5 percent increase in hydraulic resistance compared to the LOPAR assembly. Westinghouse has performed hydraulic tests at its fuel assembly test system facility to evaluate the hydraulic effects of the OFA-LOPAR mixed core. The tests were performed with a side-by-side OFA and LOPAR fuel assembly arrangement under hydraulic flow conditions approximating the reactor conditions. The results show that they are hydraulically compatible with the pressure drops within 3.5 percent of each other.

The thermal hydraulic analysis of the mixed core is performed using the same methods described in the FSAR for the 15X15 LOPAR fuel except that a Westinghouse critical heat flux (CHF) correlation designated WRB-1 is used for the OFA and the Westinghouse W-3 L-grid CHF correlation is used for the LOPAR fuel. The staff evaluation of the thermal hydraulic analysis is summarized in the following.

(a) The WRB-1 correlation (Ref. 14) was approved for the 17X17 OFA, and 17X17 and 15X15 standard LOPAR fuel assemblies with DNBR limit of 1.17 for R-grid. No CHF test data is available for the 15X15 OFA and, therefore, the application of the WRB-1 correlation to the 15X15 OFA is of concern. In response to staff questions during the D. C. Cook Unit 1, Cycle 8 reload review, W provided the 14X14 OFA CHF test data and additional proprietary information regarding the design of the 15X15 OFA. The 15X15 OFA design is virtually identical to the 15X15 R-grid design. A scaling technique was used in the 15X15 OFA grid design to ensure that the DNB performance is not affected by the OFA grid. This scaling technique has also been used for the design of the 17X17 and 14X14 OFA grids. In order to evaluate the effect of the geometry change on the accuracy of the WRB-1 correlation, W also performed a statistical analysis using the T-test and F-test for the 17X17 standard/OFA data and the 14X14 standard/OFA data. These tests are discussed in Ref. 3. The results show that the null hypothesis, the WRB-1 correlation predicts the departure from nucleate boiling (DNB) behavior of the OFA geometry with the same accuracy as the standard R-grid geometry, cannot be rejected at a 5 percent significance level. For the case where the F-test rejects the null hypothesis, the OFA data have an appreciably lower variance which is indicative of better correlation accuracy. Therefore, even though no 15X15 OFA CHF data is available, the statistical analysis performed by W has provided the basis for the applicability of the WRB-1 correlation on the 15X15 OFA.

(b) The thermal hydraulic analysis of a transitional mixed core has been previously reviewed by the staff (Ref. 15) and approved with a condition requiring a penalty on departure from nucleate boiling ratio (DNBR) to account for the uncertainty associated with the interbundle cross-flow in the mixed core.

The licensee has performed an analysis to determine the required penalty factor in the same manner approved for the 17X17 OFA/LOPAR mixed core analysis. The result shows that a 3 percent penalty is required on the OFA for the transitional mixed core. The penalty will not be required for the full core OFA fuel.

(c) The W WABA poison rod design is described in WCAP-10021, Revision 1 (Ref.9) which has been approved by the staff. In order to ensure no violation of the total calculated core bypass flow limit, the total number of WABA rods in the core should be less than the upper limit established in Table 7.2 of WCAP-10021, Revision 1. The licensee has indicated that a total of 160 WABA rods will be used in the Turkey Point Unit 3 Cycle 9 core. This number is far below the allowed limit and is, therefore, acceptable. For other reload cores, the number of WABA rods will be required to be within the allowed limit.

(d) Using the approved method for rod bow penalty calculation described in the staff review of WCAP-8691 (Ref. 16), the licensee indicated that the maximum rod bow penalty is 14 percent of DNBR corresponding to 85 percent gap closure. The staff independent calculation using the approved interim rod bow method (Ref. 17) with the revised rod bow coefficients (Ref. 18) has determined a gap closure of 85.7 percent at 33,000 MWD/MTU. Because the physical burndown effect at higher burnup is greater than the rod bowing effect which would be calculated based on the amount of bow predicted at those burnups, the 33,000 MWD/MTU represents the maximum burnup of concern for rod bow penalty calculation.

(e) For the LOPAR fuel, DNBR is calculated with the W-3 L-grid CHF correlation with the design minimum DNBR limit of 1.30. The value is 4.8 percent higher than the allowable DNBR limit of 1.24 derived from the 15X15 L-grid CHF test data. The analysis contains an inherent DNBR margin of 18.0 percent resulting from the use of conservative values of thermal diffusion coefficient and pitch reduction, the use of a conservative fuel densification model (Ref. 19) and the difference in the design and allowable DNBR limits. This DNBR margin is more than enough to compensate for the rod bow penalty of 14.9 percent.

For the 15X15 OFA fuel, a plant-specific safety analysis DNBR limit of 1.56 is used in the thermal hydraulic analysis. The safety analysis DNBR limit has a 25 percent DNBR margin compared with the DNBR limit of 1.17 for the WRB-1 CHF correlation. This 25 percent margin is more than enough to account for the rod bow penalty of 14.9 percent, the transitional mixed core penalty of 3 percent and the small uncertainty associated with the application of the WRB-1 correlation on the 15X15 OFA fuel.

(f) Based on the aforementioned evaluation, we have concluded that the use of the 15X15 OFA fuel and the WABA rods in the Turkey

Plant reloads is acceptable with the condition that the total number of WABA rods cannot exceed the upper limit imposed in Table 7.2 of WCAP-10021, Revision 1.

5.0 ACCIDENT-AND-TRANSIENT-EVALUATION

The accidents analyzed in the FSAR which could potentially be affected by the OFA design were reviewed. Since the physics characteristics of the OFA design fall into the normal range of variations seen from cycle-to-cycle, as discussed in Section 3.0, these do not lead to a need for a reevaluation of the accidents and transients.

However, the 15X15 OFA guide thimbles are similar to their counterparts in the LOPAR fuel assemblies except for 13 mil ID and OD reduction in the guide thimble above the dashpot. Due to the reduced clearance, the shutdown and control rod drop time to the dashpot for accident analyses has been determined to increase from 1.8 seconds for the LOPAR assembly to 2.4 seconds for the OFA. This increase could affect the "fast" transients for which the protection system trips the reactor within a few seconds.

An evaluation of the effect of rod drop time showed that all accidents and transients except the loss of flow, locked rotor and rod ejection are insignificantly affected by the increased rod drop time. These three accidents were reanalyzed to account for the increased rod drop time.

For the loss of reactor coolant flow accident with the 2.4 second scram time, the flow coastdown, nuclear power, heat flux and DNBR ratio versus time curves were very similar to the case with the 1.8 seconds scram time. The minimum DNBR of approximately 1.74 occurred at 3.6 seconds. This is greater than the DNBR limit of 1.56 used for safety analysis. Acceptability of the 1.56 NDBR limit is discussed in Section 4.0, item e. This result indicates no fuel failure is expected for the loss of flow accident.

The locked rotor was reanalyzed and the figures for core flow coastdown, nuclear power, reactor coolant pressure and fuel clad temperature were similar to the previous ones. Less than 10 percent of the fuel rods exhibited a DNBR less than 1.56. The peak clad temperature was 1953°F, well below any clad temperature which could be associated with a loss of coolable geometry for the core. The fuel which has a DNBR less than the limit (1.56) is assumed to fail. Site boundary doses are calculated on the basis of 10% failed fuel. This has been found acceptable in previous evaluations for the Turkey Point reactors.

When the rod ejection accident was reanalyzed the changes in the maximum fuel centerline temperature, clad average temperature, fuel enthalpy and fuel centerline melt were very small, as can be seen from Table 1. The maximum fuel enthalpy remains below 200 cal/gm, which is the Westinghouse criterion for irradiated fuel (225 cal/gm) for unirradiated fuel. The applicable NRC criterion is 280 cal/gram as defined in Regulatory Guide 1.77.

The results of the reanalysis for all three accidents thus showed that the safety limits and applicable criteria are satisfied with OFA. In addition, neither the licensee nor the NRC staff could identify any aspects of the OFA/WABA design or change in rod drop time which would create the probability of a new or different accident from any accident previously identified. We, therefore, find the OFA/WABA design and increased rod drop time acceptable.

6.0 TECHNICAL SPECIFICATIONS

The Technical Specification changes proposed for this amendment involve:

(a) Pages 3.2-2, B3.2-2

This change permits an increase in the shutdown and control rod drop time. It is acceptable, as discussed in Section 5.

(b) Page 5.2-1

This change permits the use of WABA rods. It is acceptable, as discussed in Section 2.6.

(c) Pages B2.1-1, B2.1-2, B2.3-2, B2.3-3, B3.1-1, B3.2-3 and B3.2-8.

The Technical Specification Bases on these pages have been changed by removing the DNBR limit specifically for the W-3 correlation. These changes are made to allow the use of the WRB-1 correlation

DNBR limit for the OFA fuel. Since the 15X15 OFA fuel is acceptable for the Turkey Point plants, as discussed in Section 4.0, the Technical Specification changes are acceptable.

7.0 SIGNIFICANT HAZARDS CONSIDERATION COMMENTS

These proposed amendments were noticed on July 20, 1983 (48 FR 33080) and no petition for leave to intervene or significant hazards consideration comments were received pursuant to that notice. However, a petition for leave to intervene and comments were received on separate amendment requests, which were noticed on October 7, 1983 (48 FR 45862), relating to different aspects of the core reload design. Some of these comments and concerns were relevant to the present amendments. Since these amendments had not yet issued, the staff, in its discretion, has chosen to address the comments relevant to these amendments. The comments and concerns were received from the Center for Nuclear Responsibility and Ms. Joette Lorion.

Concerns were expressed that a newly designed fuel assembly in conjunction with a new type of rod which has never been installed or tested under field operating conditions will be used and, in as far as the commenters could determine, the staff has not published a proposed safety evaluation report.

These concerns have been addressed in Sections 2.0, 3.0 and 4.0 of this safety evaluation. The results of our evaluation of the mechanical, physics and thermal hydraulic characteristics indicate that: (1) the

OFA/WABA reload core is not significantly different from those previously found acceptable at Turkey Point, (2) there are no significant changes to the acceptance criteria for the Technical Specifications, and (3) the analytical methods applicable to the OFA/WABA reload core are not significantly changed and we have previously found them acceptable.

Concerns were expressed that these amendments would increase the rod drop time from 1.8 to 2.4 seconds (a 33% increase in rod drop time) and that the increase would significantly and adversely reduce the safety margin and create the possibility for, or probability of, a new or different kind of accident, or an accident whose occurrence or consequences have not been analyzed, or which may increase the probability of an accident previously analyzed. The Center for Nuclear Responsibility and Joette Lorion also contend that Commission's tentative conclusion that safety limits "are met" is not supported by any evidence.

These concerns are addressed in Section 5.0 of this Safety Evaluation. The results of our evaluation of the design basis accidents or transients and reanalysis of the events affected by the increase in rod drop time indicate that the increase does not significantly and adversely reduce the safety margin or create the possibility for or probability of a new or different kind of accident or any accident whose occurrence or consequences have not been analyzed or significantly increase the probability of an accident previously analyzed.

8.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

Due to the unusual circumstances surrounding this amendment (i.e, the filing of a petition for leave to intervene on separate proposed amendments and substantive comments relating to the present amendments substantially after the 30 day comment period, but before issuance of the present amendments) the staff, in its discretion, has made a final no significant hazards consideration determination.

The Commission has provided guidance concerning the application of the standards for determing whether license amendments involve no significant hazards considerations by providing certain examples (48 FR 14870). Example (iii) of amendments which were not likely to involve significant hazards consideration are changes resulting from nuclear reactor reloading involving no fuel assemblies significantly different from those previously found acceptable at the facility in question, where no significant changes are made to the acceptance criteria for the Technical Specifications, the analytical methods used are not significantly changed and the NRC has previously found the methods acceptable.

These amendments are similar to this example in that the W 15X15 OFA fuel is very simlar to the W 15X15 standard low parasitic (LOPAR) fuel design currently used at Turkey Point Units 3 and 4. The physics characteristics for the OFA fuel are only slightly different from those of the LOPAR. These differences are within the normal range of variations seen from cycle to cycle. Furthermore, the 15X15 Zircaloy grid design is similar to the W 17X17 grid design, which is described in WCAP-9500-A. This report has been reviewed and approved by the NRC staff. The design criteria and evaluation methods used in WCAP-9500-A were also used for th 15X15 OFA and approved by the staff for the D. C. Cook Unit 1, Cycle 8, reload. D. C. Cook used identical fuel as that used in Turkey Point. The physical change introduced by 15X15 optimized fuel assembly (OFA) design is the use of 5 intermediate Zircaloy grids replacing 5 intermediate Inconel grids in the LOPAR fuel. The Zircaloy grids have thicker and wider straps than the Inconel grids in order to closely match the Inconel grid stength and have a slight increase in the hydraulic resistance. However, the pressure drops in LOPAR and OFA assemblies are within 3.5 percent of each other.

Due to its similarities to the LOPAR fuel as discussed above and detailed in Sections 2.0, 3.0, 4.0 and 5.0 of this evaluation, the use of the OFA fuel does not involve a significant increase in the probability or consequences of an accident previously evaulated. The use of the OFA fuel does not create the probability of a new or different accident from any accident previously evaluated. See discussion in Section 5.0. The OFA fuel is very similar to the LOPAR fuel and its use does not involve a significant reduction in a margin of safety. This is discussed in Sections 2.0, 3.0, 4.0 and 5.0 of this report.

The use of Wet Annular Burnable Absorber (WABA) rods has been generically reviewed and approved by the NRC staff for use in W core designs. The

WABA rod design consists of aluminum oxide and boron carbide burnable absorber material encapsulated within two concentric Zircaloy tubes. These rods contain no fuel. The only safety concern is related to assuring that flow through the WABA rods does not result in excess flow bypassing the core. To assure adequate flow through the core, an upper bound for the number of WABA rods was computed and identified in Table 7.2 of WCAP 10021, REV. 1. Turkey Point 3, Cycle 9, will use 160 WABA rods which is substantially below the limit. All future reloads are required to be within the allowable limits.

The use of WABA rods does not: 1) involve a significant increase in the probability or consequences of an accident previously evaluated in that the rods contain no fuel and the total core bypass flow is well within the limits of our generic review which approved the use of WABA rods with W core designs as indicated above and in Sections 2.0, 4.0 and 5.0, of this evaluation, 2) create the probability of a new or different accident from any accident previously evaluated, as discussed in Section 5.0 or, 3) involve a significant reduction in a margin of safety because the total number of WABA rods used is substantially below the limit established for the current reload and all future reloads are required to be within the allowable number of WABA rods as discussed above and in Sections 2.0, 4.0 and 5.0 of this evaluation.

The analytical and calculational methods used in addressing the mechanical design, physics design and thermal hydraulic evaluation for the OFA/WABA core have been previously reviewed and approved by the NRC staff. These are identified in this Safety Evaluation.

The analytical and calculational methods used, as discussed above and identified in Sections 2.0, 3.0, 4.0 and 5.0 of this report, have been previously reviewed and approved by the NRC staff and do not 1) involve a significant increase in the probability or consequences of an accident previously evaluated, 2) create the probability of a new or different accident from any accident previously evaluated or, 3) involve a significant reduction in a margin of safety.

The design basis accidents analyzed in the Final Safety Analysis Report which could potentially be affected by the OFA/WABA core design were reviewed. Since the physics characteristics of the OFA/WABA design fall into the normal range of variations seen from cycle to cycle, these do not lead to a need for reevaluation of the accidents and transients. However, the shutdown and control rod drop time is increased from 1.8 seconds to 2.4 seconds. This could affect the accidents or transients which require the protection system to trip the reactor within a few seconds. The only accidents or transients affected by the increase in the rod drop time are the loss of flow, locked rotor and rod ejection. These accidents were reanalyzed to account for the increased rod drop time. The FSAR design basis and the acceptance criteria specified in the Standard Review Plan were used to determine the acceptability of the reanalysis. The results were within the limits of the FSAR design basis and criteria specified in the Standard Review Plan, therefore resulting in no significant changes in the results or consequences of these accidents or transients.

The shutdown and control rod drop time increase does not: 1) increase the probability or consequences of an accident previously evaluated based on the reanalysis discussed above and in Section 5.0 of this evaluation, 2) create the probability of a new or different accident from any accident previously identified, as discussed in Section 5.0, or, 3) involve a significant reduction in a margin of safety because the results of the reanalysis indicate no fuel failure for the loss of reactor coolant flow, less than 10 percent fuel failure for the locked rotor, and the maximum fuel enthalpy is below the 280 cal/gm for the rod ejection accident. All of the results of the reanalysis are still within the limits of the FSAR design basis and criteria specified in the Standard Review Plan as discussed above and detailed in Section 5.0 of this evaluation.

Based on our review of the licensee's submittal, as described above and in our safety evaluation, we have made a final determination that the amendments do not 1) involve a significant increase in the probability or consequences of an accident previously evaluated, 2) create the probability of a new or different accident from any accident previously evaluated, or 3) involve a significant reduction in a margin of safety; and therefore, do not involve a significant hazards consideration.

9.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

10.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) these amendments do not involve significant hazards considerations, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: December 9, 1983

Principal Contributors:

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REFERENCES

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2. R. L. Tedesco (NRC) letter to T. M. Anderson (W), "Reference Core Report 17X17 Optimized Fuel Assembly," May 22, 1981.

3. R. F. Hering (AEP) letter to H. R. Denton (NRC), August 31, 1983.

4. WCAP-8377(P)/WCAP-8381 (NP) Accepted by letter dated February 14, 1975 - D. B.Vassallo (NRC) to C. Echeldinger (W).

5. WCAP-8720 Addendum 2 (P) Approval letter C. O. Thomas (NRC) to E. P. Rahe Jr., December 9, 1983.

6. WCAP-8720, WCAP-8720, Addendum 1, WCAP-8720 Accepted by letter March 27, 1980-J. Stolz (NRC) to T. Anderson (W), WCAP-8720, Addendum 1, Accepted by letter dated July 20, 1982 - Harold Bernard (NRC) to E. P. Rahe (W).

7. WCAP-9220(P)/922(NP), Accepted by letter datead December 1, 1981-J. R. Miller (NRC) to E. P. Rahe (W).

8. "Mechanical Fracture Evaluation of Reactor Coolant Pipe Containing a Postulated Circumferential Through-Wall Crack," WCAP-9558, Revision 2, May 1982; "Tensile and Toughness Properties of Primary Piping Weld Metal for Use in Mechanistic Fracture Evaluation," WCAP-9787, May 1981.

9.& 10. WCAP-10021(P)/WCAP-10377(NP), Accepted by letter dated August 9, 1983 C. O. Thomas (NRC) to E. P. Rahe (W).

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13. WCAP-9273, Westinghouse Reload Safety Evaluation Methodology (NP) March 1978.

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15. WCAP-9500, Accepted by letter datead May 22, 1981 - R. L. Tedesco (NRC) to T. M. Anderson (W). WCAP-9401(P)/9402(NP), Accepted by letter dated May 7, 1981, R. L. Tedesco (NRC) to T. M. Anderson (W) Supplemental Acceptance Number 1 on November 12, 1982, by letter C. O. Thomas (NRC) to E. P. Rahe (W) Supplemental Acceptance Number 2 letter dated January 24, 1983, C. O. Thomas (NRC) to E. P. Rahe (W).

16. WCAP-8691, Accepted by letter dated April 5, 1979, letter from J. Stolz (NRC) to T. M. Anderson (W).

17. Memorandum from D. Ross and D. Eisenhut (NRC) to D. Vassallo and K. Goller, "Revised Interim Safety Evaluation Report on the Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors, "February 16, 1977. (In PDR)

18. Memorandum from R. O. Meyer to D. F. Ross, "Revised Coefficients for Interim Rod Bowing Analysis, "March 2, 1978. (Proprietary CE, B&W, W)

19. WCAP-8218(P)/8219(NP), Accepted by letter dated June 25, 1974 by letter from D. B. Vassallo (NRC) to Romano Salvatori (W).

U. S. NUCLEAR REGULATORY COMMISSION

FLORIDA POWER AND LIGHT COMPANY

DOCKET NOS. 50-250 AND 50-251

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITIES

OPERATING LICENSES

AND FINAL DETERMINATION OF NO SIGNIFICANT

HAZARDS CONSIDERATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No.98 to Facility Operating License No. DPR-31, and Amendment No. 92 to Facility Operating License No. DPR-41 issued to Florida Power and Light Company (the licensee), which revised Technical Specifications for operation of Turkey Point Plant, Unit Nos. 3 and 4 (the facilities) located in Dade County, Florida. The amendments are effective as of the date of issuance for Unit 3 and startup of Cycle 10 for Unit 4.

The application for these amendments comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commisison has made appropirate findings as requied by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the these license amendments.

Notice of Consideration of Issuance of Amendments and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER (48 FR 33080) on July 20, 1983. No significant hazards considerations comments have been received on this action, but comments relevant to this action have been received on a related amendment (48 FR 45862, October 7, 1983).

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards consideration is involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that these amendments involve no significant hazards consideration. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, these amendments have been issued and made immediately effective for Unit 3 and startup Cycle 10 for Unit 4.

The Commission has determined that the issuance of the amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendments.

For further details with respect to the action see (1) the application for amendments dated June 3, 1983, as supplemented November 16, 1983, (2) Amendment Nos. 98 and 92 to Facilities Operating License Nos. DPR-31 and DPR-41 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C.., and at the Environmental and

Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 9th day of December , 1983.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing