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

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. ROD DROP TIME
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. INOPERABLE CONTROL RODS
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-

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* 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(Δq) 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.

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

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 $\leq$ 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...

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

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.

$F_Q(Z)$, Heat Flux Hot Channel Factor, 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, Engineering Heat Flux Hot Channel Factor, 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.

$F_{\Delta H}^N$, Nuclear Enthalpy Rise Hot Channel Factor, 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 $\pm$ 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 $\bar{R}$, 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.