MAY 27 1979

Docket No.: 50-335

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Dr. Robert E. Uhrig
Vice President
Florida Power & Light Company
Advanced Systems & Technology
P. O. Box 529100
Miami, Florida 33152

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 32 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. The amendment consists of changes to the license and its appended Technical Specifications in response to your requests of November 9, 1978 and February 22, 1979, as supplemented April 30 and May 1, 10, 18, 22 and 23, 1979.

This amendment revises the Technical Specifications to incorporate changes resulting from Cycle 3 reload fuel and deletes a license requirement for neutron shielding which has been satisfactorily completed.

On March 28, 1979, Three Mile Island, Unit No. 2 (TMI-2) experienced core damage which resulted from a series of events which were initiated by a Loss of Feedwater Event and apparently compounded by operational errors. We have determined that several aspects of this accident have generic applicability to all light water power reactor facilities such as St. Lucie Plant, Unit No. 1. To identify corrective actions to be taken by all licensees, I&E bulletins have been issued. The particular bulletin that applies to St. Lucie Plant, Unit No. 1 is Bulletin No. 79-06B.

7907090/77

You provided the Florida Power & Light Company response to Bulletin 79-06B by letter to J. O'Reilly dated April 24, 1979. Our evaluation of the response and subsequent revision dated May 24, 1979 indicates that the actions taken by Florida Power & Light Company demonstrate understanding of the salient concerns arising from the TMI-2 accident as it relates to St. Lucie Plant, Unit No. 1 operations. Florida Power & Light Company actions also provide added assurance for the protection of the health and safety of the public during plant operation. A separate safety evaluation will be issued to document our review of the Florida Power & Light Company response to I&E Bulletin No. 79-06B and to identify certain areas where additional information or action is required.

OFFICE						
SURNAME						
DATE						

In our evaluation of your request for approval of reload Cycle 3 Technical Specifications, we have determined that additional information on sleeved guide tubes is required prior to the next refueling shutdown. Therefore, you are requested to provide an evaluation program (including the planned inspections) to determine guide tube wear experienced after the next cycle of operation with sleeved fuel assemblies. Provide this evaluation program at least 90 days prior to the Cycle 4 reload outage.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original signed by

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Enclosures:
1. Amendment No. 32 to DPR-67
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

OFFICE	ORB#4:DOR	ORB#4:DOR	C-ORB#4:DOR	OELD	AD:L&P:DOR	RSB
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DATE	5/25/79	5/25/79	5/27/79	5/25/79	5/25/79	27 May 79

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

May 27, 1979

Docket No.: 50-335

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Vice President
Florida Power & Light Company
Advanced Systems & Technology
P. O. Box 529100
Miami, Florida 33152

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Operating Reactors Branch #4
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1. Amendment No. 32 to DPR-67
2. Safety Evaluation
3. Notice

cc w/enclosures: See next page

Florida Power & Light Company

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Bureau of Intergovernmental
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Tallahassee, Florida 32304

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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. DPR-67

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

A. The applications for amendment by Florida Power & Light Company (the licensee) dated November 9, 1978 and February 22, 1979, as supplemented April 30 and May 1, 10, 18, 22 and 23, 1979, comply 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 applications, 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.

184

7907090

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by the following changes to Facility Operating License No. DPR-67:

A. Revise paragraph 2.C.(2) in its entirety to read as follows:

Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 32 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

B. Delete in its entirety condition D of Enclosure 1 appended to the license.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 27, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 32

FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

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STAGGERED TEST BASIS

1.21 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.22 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

AXIAL SHAPE INDEX

1.23 The AXIAL SHAPE INDEX (Y_E) is the power level detected by the lower excore nuclear instrument detectors (L) less the power level detected by the upper excore nuclear instrument detectors (U) divided by the sum of these power levels. The AXIAL SHAPE INDEX (Y_I) used for the trip and pretrip signals in the reactor protection system is the above value (Y_E) modified by an appropriate multiplier (A) and a constant (B) to determine the true core axial power distribution for that channel.

$$Y_E = \frac{L-U}{L+U} \qquad Y_I = AY_E + B$$

UNRODDED PLANAR RADIAL PEAKING FACTOR - F_{xy}

1.24 The UNRODDED PLANAR RADIAL PEAKING FACTOR is the maximum ratio of the peak to average power density of the individual fuel rods in any of the unrodded horizontal planes, excluding tilt.

SHIELD BUILDING INTEGRITY

1.25 SHIELD BUILDING INTEGRITY shall exist when:

1.25.1 Each door is closed except when the access opening is being used for normal transit entry and exit, and

1.25.2 The shield building ventilation system is OPERABLE.

REACTOR TRIP SYSTEM RESPONSE TIME

1.26 The REACTOR TRIP SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until electrical power is interrupted to the CEA drive mechanism.

ENGINEERED SAFETY FEATURE RESPONSE TIME

1.27 The ENGINEERED SAFETY FEATURE RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays where applicable.

PHYSICS TESTS

1.28 PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14.0 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

UNRODDED INTEGRATED RADIAL PEAKING FACTOR - F_r

1.29 The UNRODDED INTEGRATED RADIAL PEAKING FACTOR is the ratio of the peak pin power to the average pin power in an unrodded core, excluding tilt.

LOAD FOLLOW OPERATION

1.30 LOAD FOLLOW OPERATION shall be daily power level changes of more than 10% of RATED THERMAL POWER or daily insertion of CEAs below the Long Term Insertion Limit.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR TRIP SETPOINTS

2.2.1 The reactor protective instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2-1.

APPLICABILITY: AS SHOWN FOR EACH CHANNEL IN TABLE 3.3-1.

ACTION:

With a reactor protective instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1.1 until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

TABLE 2.2-1

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Manual Reactor Trip	Not Applicable	Not Applicable
2. Power Level - High (1)		
Four Reactor Coolant Pumps Operating	≤ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of ≤ 107.0% of RATED THERMAL POWER.	≤ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RATED THERMAL POWER and a maximum of ≤ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	≥ 95% of design reactor coolant flow with 4 pumps operating*	≥ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	≤ 2400 psia	≤ 2400 psia
5. Containment Pressure - High	≤ 3.3 psig	≤ 3.3 psig
6. Steam Generator Pressure - Low (2)	≥ 500 psia	≥ 500 psia
7. Steam Generator Water Level -Low	≥ 37.0% Water Level - each steam generator	≥ 37.0% Water Level - each steam generator
8. Local Power Density - High (3)	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2	Trip set point adjusted to not exceed the limit lines of Figures 2.2-1 and 2.2-2.

*Design reactor coolant flow with 4 pumps operating is 370,000 gpm.

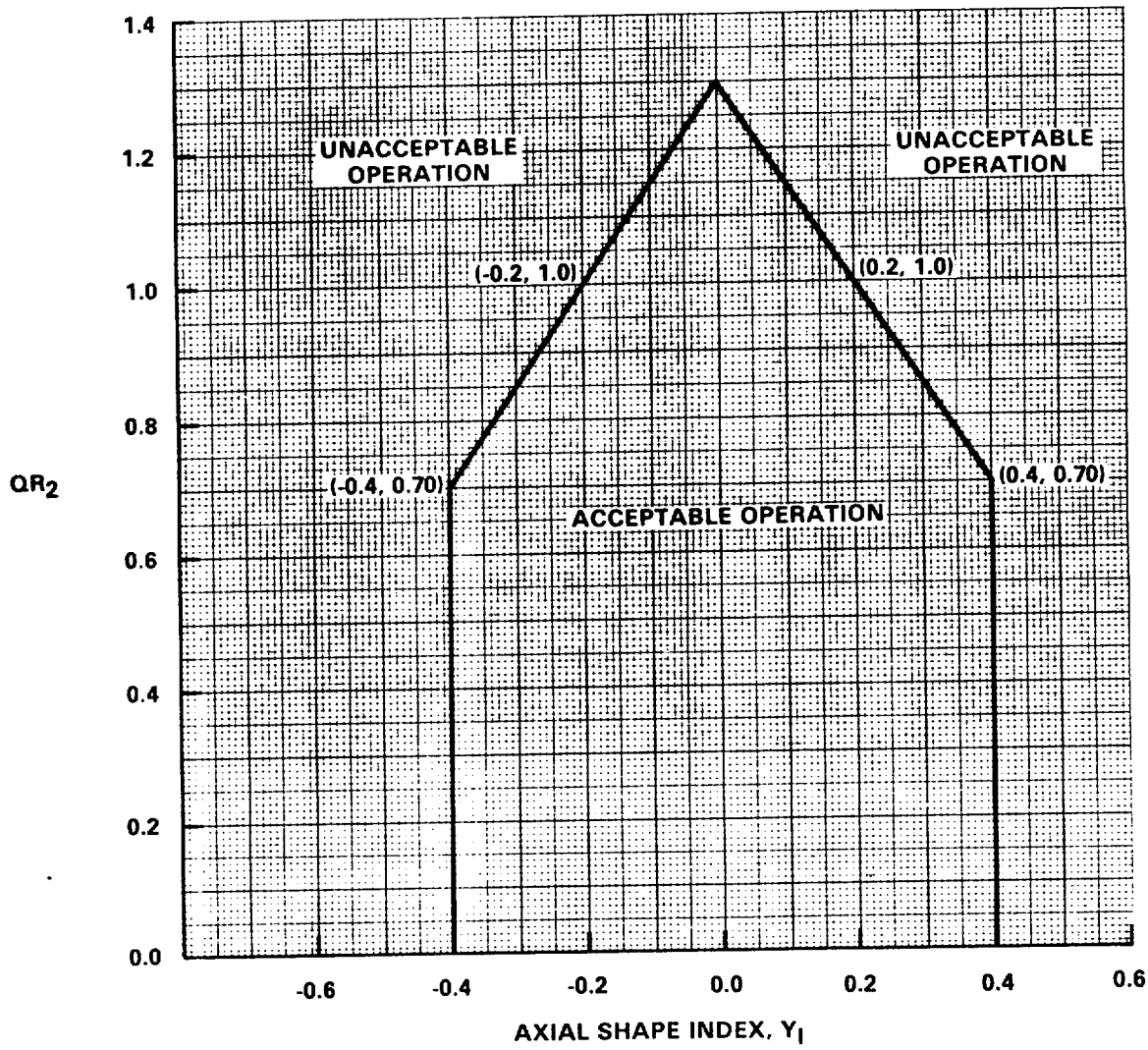

FIGURE 2.2-2
Local Power Density-High Trip Setpoint Part 2 (QR_2 Versus Y_I)

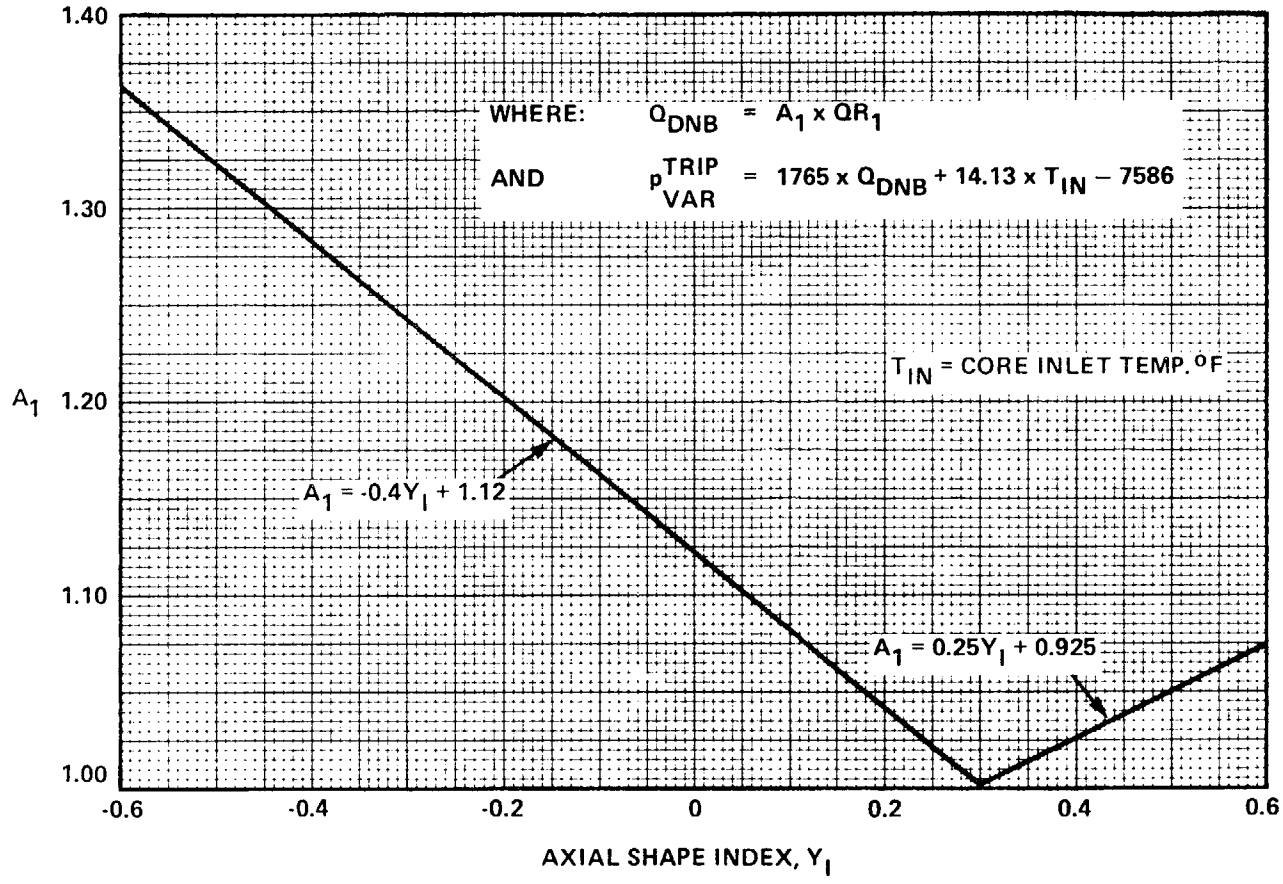

FIGURE 2.2-3
Thermal Margin/Low Pressure Trip Setpoint
Part 1 (Y_I Versus A_1)

Reactor Coolant Flow-Low (Continued)

reactor coolant pumps are taken out of service. The low-flow trip setpoints and Allowable Values for the various reactor coolant pump combinations have been derived in consideration of instrument errors and response times of equipment involved to maintain the DNBR above 1.30 under normal operation and expected transients. For reactor operation with only two or three reactor coolant pumps operating, the Reactor Coolant Flow-Low trip setpoints, the Power Level-High trip setpoints, and the Thermal Margin/Low Pressure trip setpoints are automatically changed when the pump condition selector switch is manually set to the desired two- or three-pump position. Changing these trip setpoints during two and three pump operation prevents the minimum value of DNBR from going below 1.30 during normal operational transients and anticipated transients when only two or three reactor coolant pumps are operating.

Pressurizer Pressure-High

The Pressurizer Pressure-High trip, backed up by the pressurizer code safety valves and main steam line safety valves, provides reactor coolant system protection against overpressurization in the event of loss of load without reactor trip. This trip's setpoint is 100 psi below the nominal lift setting (2500 psia) of the pressurizer code safety valves and its concurrent operation with the power-operated relief valves avoids the undesirable operation of the pressurizer code safety valves.

Containment Pressure-High

The Containment Pressure-High trip provides assurance that a reactor trip in initiated concurrently with a safety injection.

Steam Generator Pressure-Low

The Steam Generator Pressure-Low trip provides protection against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant. The setting of 500 psia is sufficiently below the full-load operating point of 800 psig so as not

Steam Generator Pressure-Low (Continued)

to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow. This setting was used with an uncertainty factor of ± 22 psi in the accident analyses.

Steam Generator Water Level - Low

The Steam Generator Water Level-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity and assures that the design pressure of the reactor coolant system will not be exceeded due to loss of steam generator heat sink. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to provide a margin of more than 10 minutes before auxiliary feedwater is required.

Local Power Density-High

The local Power Density-High trip, functioning from AXIAL SHAPE INDEX monitoring, is provided to ensure that the peak local power density in the fuel which corresponds to fuel centerline melting will not occur as a consequence of axial power maldistributions. A reactor trip is initiated whenever the AXIAL SHAPE INDEX exceeds the allowable limits of Figure 2.2-2. The AXIAL SHAPE INDEX is calculated from the upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of THERMAL POWER level with the allowed CEA group position being inferred from the THERMAL POWER level. The trip is automatically bypassed below 15 percent power.

The maximum AZIMUTHAL POWER TILT and maximum CEA misalignment permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with the Specifications 3.1.3.5 and 3.1.3.6 is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a Power Level-High trip is assumed.

POSITION INDICATOR CHANNELS (Continued)

LIMITING CONDITION FOR OPERATION

b) The CEA group(s) with the inoperable position indicator is fully inserted, and subsequently maintained fully inserted, while maintaining the withdrawal sequence and THERMAL POWER level required by Specification 3.1.3.6 and when this CEA group reaches its fully inserted position, the "Full In" limit of the CEA with the inoperable position indicator is actuated and verifies this CEA to be fully inserted. Subsequent operation shall be within the limits of Specification 3.1.3.6.

c. With a maximum of one reed switch position indicator channel per group or one pulse counting position indicator channel per group inoperable and the CEA(s) with the inoperable position indicator channel at either its fully inserted position or fully withdrawn position, operation may continue provided:

1. The position of this CEA is verified immediately and at least once per 12 hours thereafter by its "Full In" or "Full Out" limit (as applicable),

2. The fully inserted CEA group(s) containing the inoperable position indicator channel is subsequently maintained fully inserted, and

3. Subsequent operation is within the limits of Specification 3.1.3.6.

d. With one or more pulse counting position indicator channels inoperable, operation in MODES 1 and 2 may continue for up to 24 hours provided all of the reed switch position indicator channels are OPERABLE.

SURVEILLANCE REQUIREMENTS

4.1.3.3 Each position indicator channel shall be determined to be OPERABLE by verifying the pulse counting position indicator channels and the reed switch position indicator channels agree within 4.5 inches at least once per 12 hours except during time intervals when the Deviation circuit is inoperable, then compare the pulse counting position indicator and reed switch position indicator channels at least once per 4 hours.

CEA DROP TIME

LIMITING CONDITION FOR OPERATION

3.1.3.4 The individual full length (shutdown and control) CEA drop time, from a fully withdrawn position, shall be $\leq$ 3.1 seconds from when electrical power is interrupted to the CEA drive mechanism until the CEA reaches its 90 percent insertion position with:

a. $T_{avg} \geq 515°F$, and

b. All reactor coolant pumps operating.

APPLICABILITY: MODE 3.

ACTION:

a. With the drop time of any full length CEA determined to exceed the above limit, restore the CEA drop time to within the above limit prior to proceeding to MODE 1 or 2.

b. With the CEA drop times within limits but determined at less than full reactor coolant flow, operation may proceed provided THERMAL POWER is restricted to less than or equal to the maximum THERMAL POWER level allowable for the reactor coolant pump combination operating at the time of CEA drop time determination.

SURVEILLANCE REQUIREMENTS

4.1.3.4 The CEA drop time of full length CEAs shall be demonstrated through measurement prior to reactor criticality:

a. For all CEAs following each removal of the reactor vessel head,

b. For specifically affected individual CEAs following any maintenance on or modification to the CEA drive system which could affect the drop time of those specific CEAs, and

c. At least once per 18 months.

3/4.2 POWER DISTRIBUTION LIMITS

LINEAR HEAT RATE

LIMITING CONDITION FOR OPERATION

3.2.1 The linear heat rate shall not exceed the limits shown on Figure 3.2-1.

APPLICABILITY: MODE 1.

ACTION:

With the linear heat rate exceeding its limits, as indicated by four or more coincident incore channels or by the AXIAL SHAPE INDEX outside of the power dependent control limits of Figure 3.2-2, within 15 minutes initiate corrective action to reduce the linear heat rate to within the limits and either:

a. Restore the linear heat rate to within its limits within one hour, or

b. Be in HOT STANDBY within the next 6 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The provisions of Specification 4.0.4 are not applicable.

4.2.1.2 The linear heat rate shall be determined to be within its limits by continuously monitoring the core power distribution with either the excore detector monitoring system or with the incore detector monitoring system.

4.2.1.3 Excore Detector Monitoring System - The excore detector monitoring system may be used for monitoring the core power distribution by:

a. Verifying at least once per 12 hours that the full length CEAs are withdrawn to and maintained at or beyond the Long Term Steady State Insertion Limit of Specification 3.1.3.6.

b. Verifying at least once per 31 days that the AXIAL SHAPE INDEX alarm setpoints are adjusted to within the limits shown on Figure 3.2-2.

c. Verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2, where 100 percent of maximum allowable power represents the maximum THERMAL POWER allowed by the following expression:

$$M \times N$$

where:

1. M is the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

2. N is the maximum allowable fraction of RATED THERMAL POWER as determined by the F^T_{xy} curve of Figure 3.2-3.

4.2.1.4 Incore Detector Monitoring System - The incore detector monitoring system may be used for monitoring the core power distribution by verifying that the incore detector Local Power Density alarms:

a. Are adjusted to satisfy the requirements of the core power distribution map which shall be updated at least once per 31 days of accumulated operation in MODE 1.

b. Have their alarm setpoint adjusted to less than or equal to the limits shown on Figure 3.2-1 when the following factors are appropriately included in the setting of these alarms:

1. Flux peaking augmentation factors as shown in Figure 4.2-1,

2. A measurement-calculational uncertainty factor of 1.07,*

3. An engineering uncertainty factor of 1.03,

4. A linear heat rate uncertainty factor of 1.01 due to axial fuel densification and thermal expansion, and

5. A THERMAL POWER measurement uncertainty factor of 1.02.

*An uncertainty factor of 1.10 applies when in LOAD FOLLOW OPERATION.

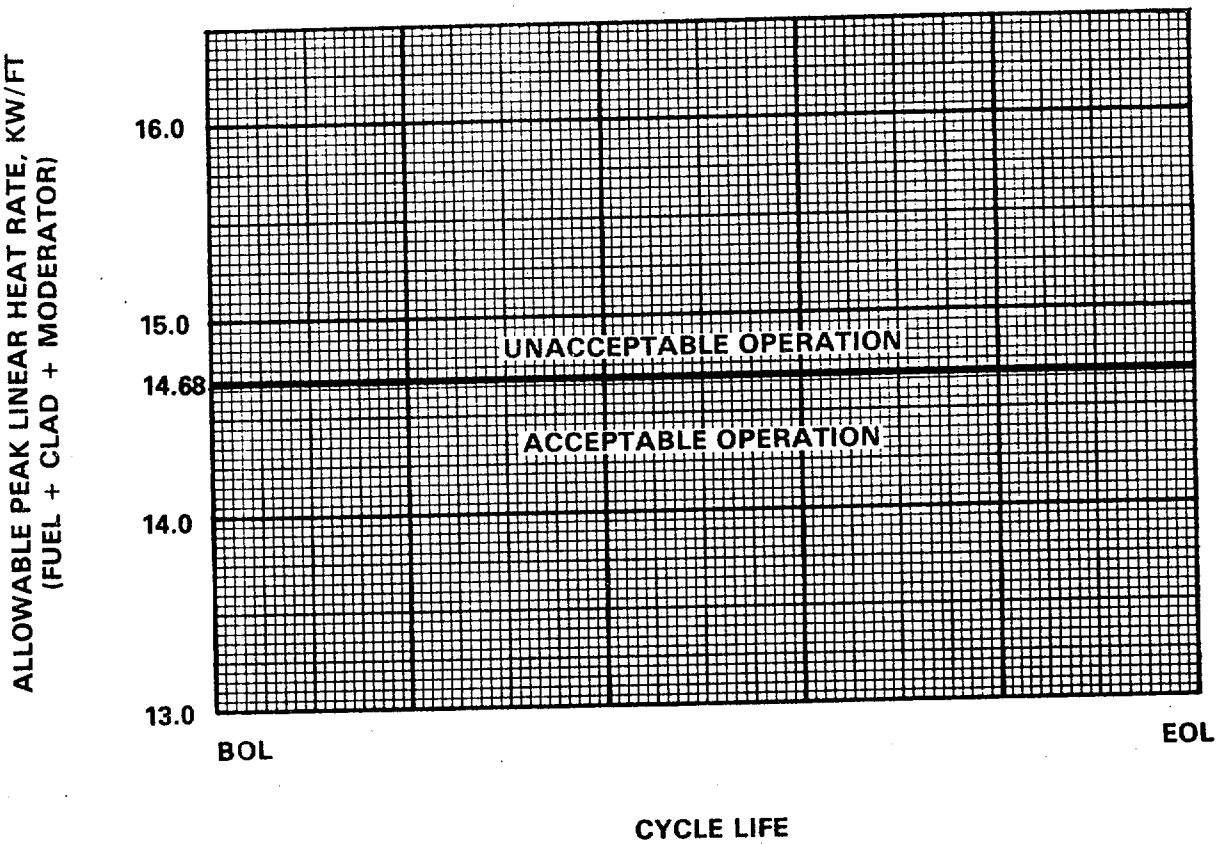

FIGURE 3.2-1 Allowable Peak Linear Heat Rate vs Burnup.

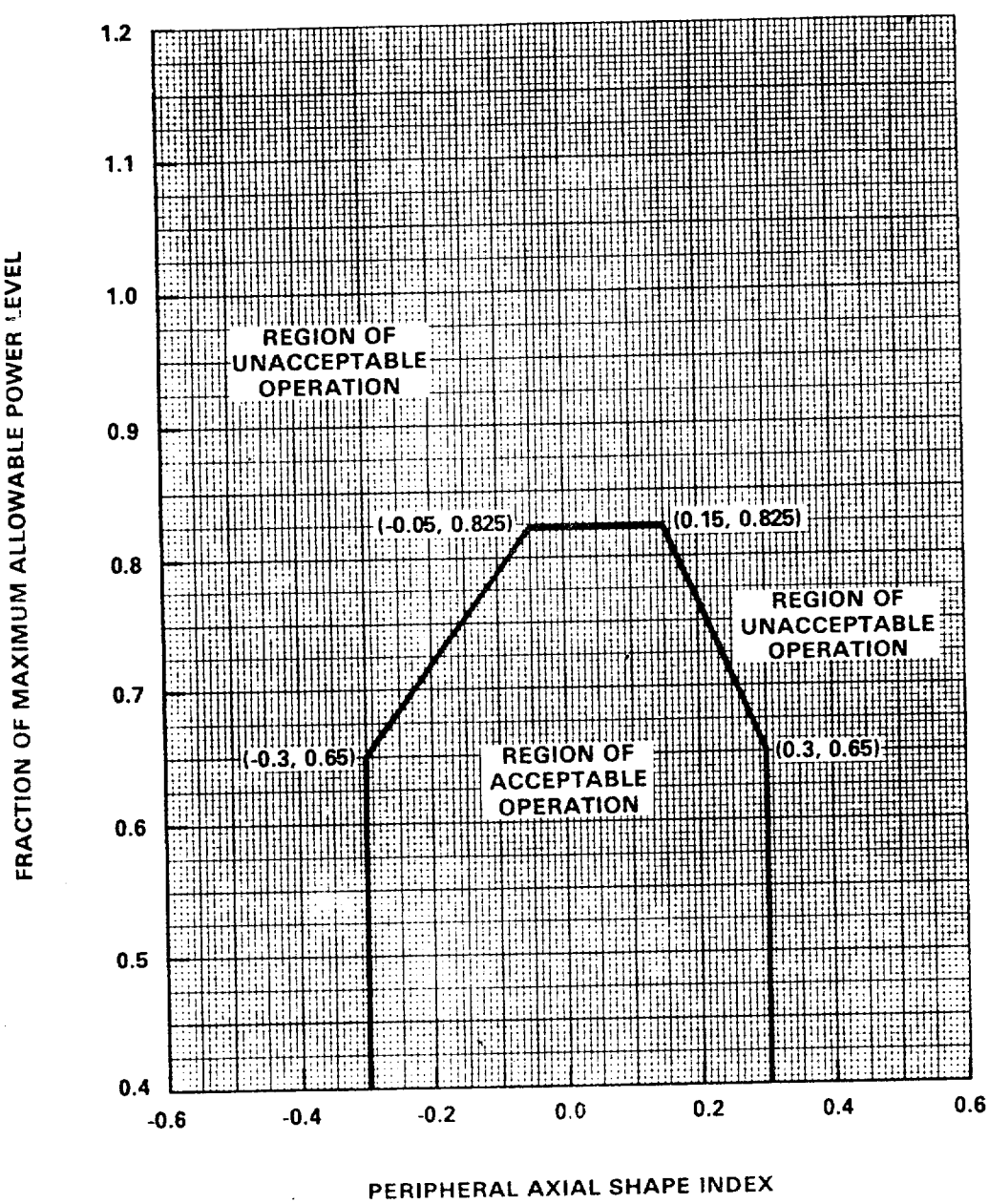

FIGURE 3.2-2
AXIAL SHAPE INDEX vs Fraction of Maximum Allowable Power Level Per Specification 4.2.1.3

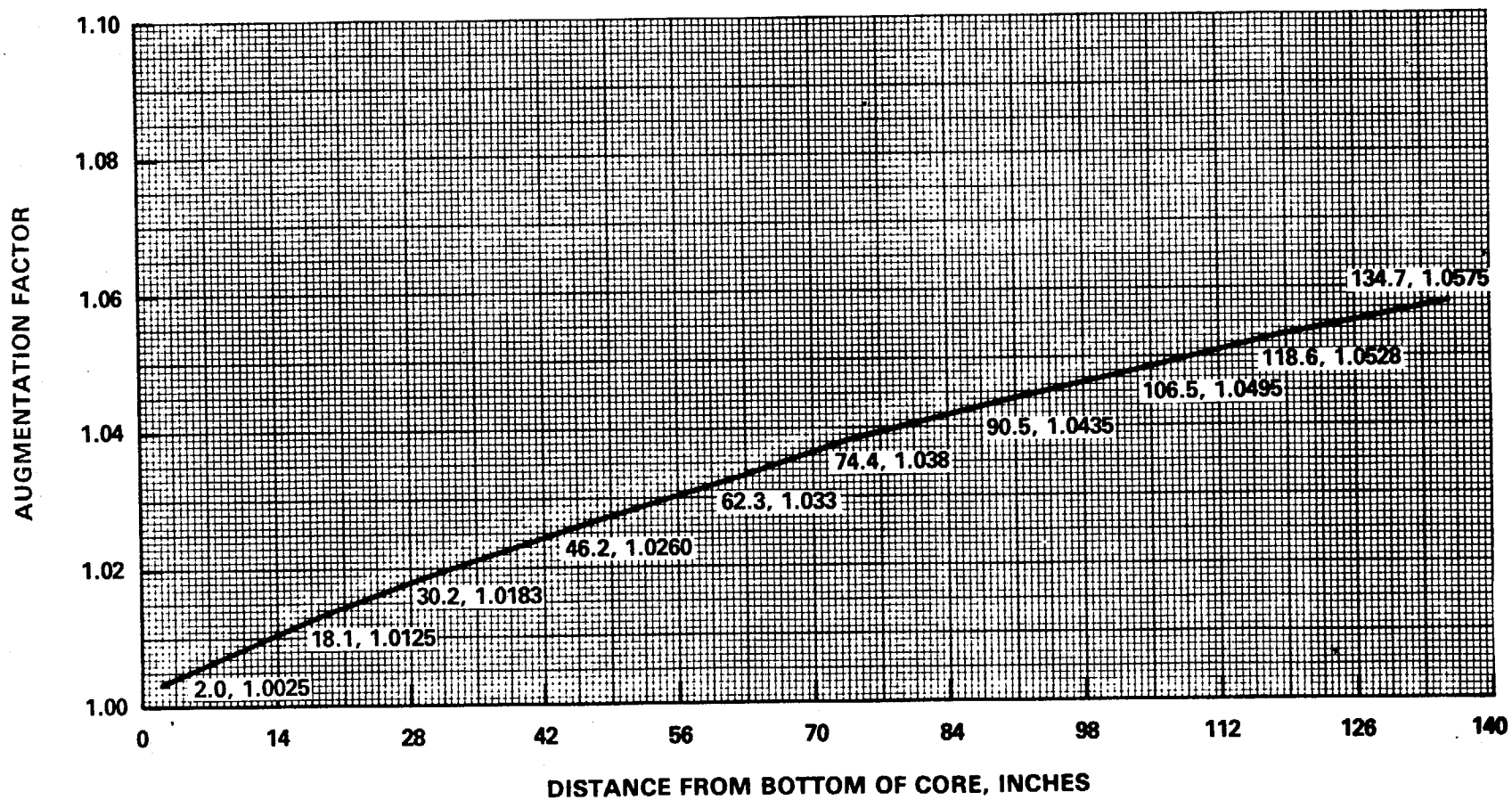

FIGURE 4.2-1
Augmentation Factors vs Distance from Bottom of Core

TOTAL PLANAR RADIAL PEAKING FACTOR - F^T_{xy}

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F^T_{xy}, defined as $F^T_{xy} = F_{xy}(1+T_q)$, shall be limited to $\leq$ 1.627.

APPLICABILITY: MODE 1*.

ACTION:

With F^T_{xy} > 1.627, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F^T_{xy} to within the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6; or

b. Be in HOT STANDBY.

SURVEILLANCE REQUIREMENTS

4.2.2.1 The provisions of Specification 4.0.4 are not applicable.

4.2.2.2 F^T_{xy} shall be calculated by the expression $F^T_{xy} = F_{xy}(1+T_q)$ when in non-LOAD FOLLOW OPERATION and by the expression $F^T_{xy} = 1.03\ F_{xy}(1+T_q)$ when in LOAD FOLLOW OPERATION. F^T_{xy} shall be determined to be within its limit at the following intervals:

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AZIMUTHAL POWER TILT (T_q) is > 0.03.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.2.3 F_{xy} shall be determined each time a calculation of F^T_{xy} is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination. This determination shall be limited to core planes between 15% and 85% of full core height and shall exclude regions influenced by grid effects.

4.2.2.4 T_q shall be determined each time a calculation of F^T_{xy} is required and the value of T_q used to determine F^T_{xy} shall be the measured value of T_q.

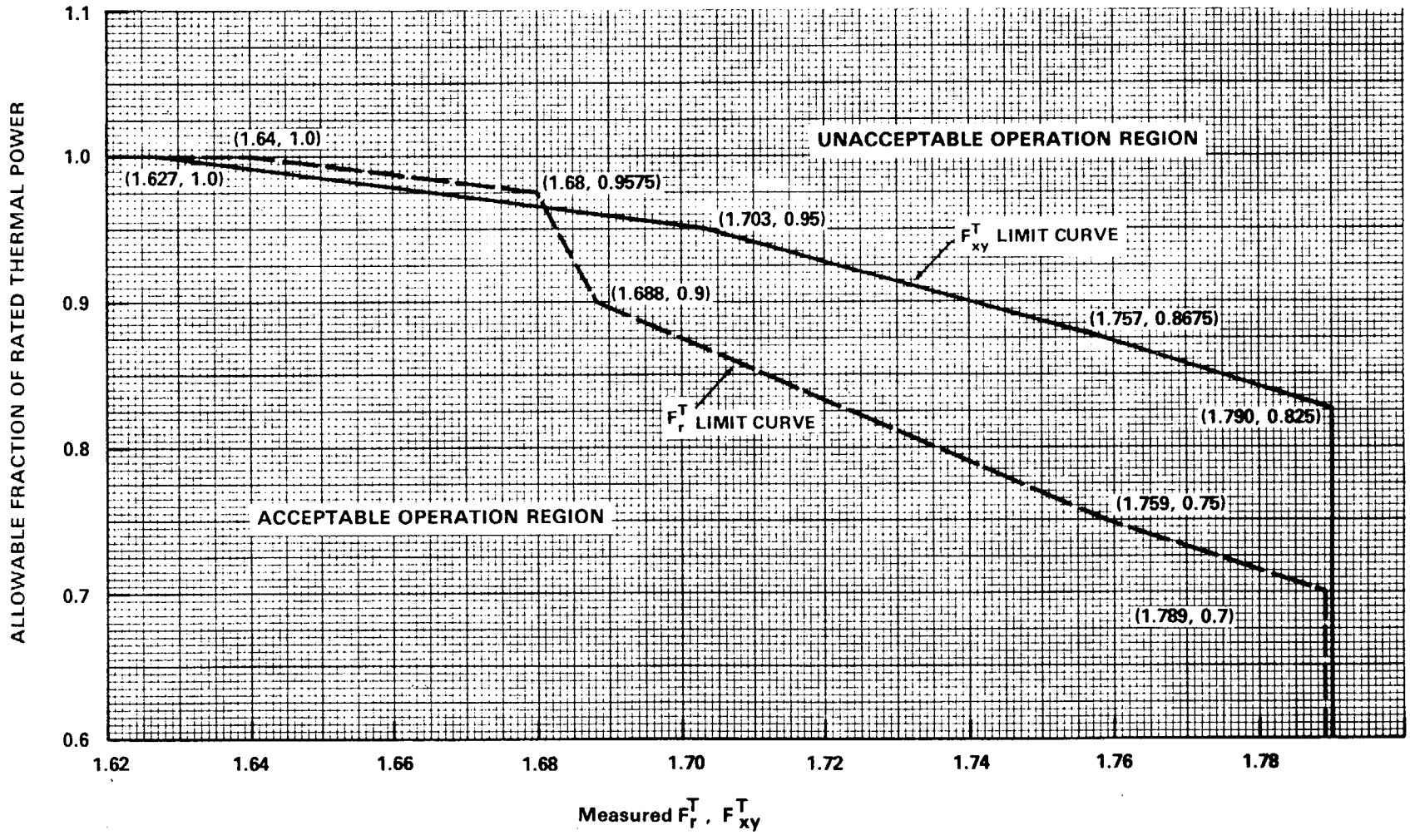

FIGURE 3.2-3
Allowable Combinations Of Thermal Power And F_r^T, F_{xy}^T

TOTAL INTEGRATED RADIAL PEAKING FACTOR - F_r^T

LIMITING CONDITION FOR OPERATION

3.2.3 The calculated value of F_r^T, defined as $F_r^T = F_r(1+T_q)$, shall be limited to $\leq$ 1.64.

APPLICABILITY: MODE 1*.

ACTION:

With F_r^T > 1.64, within 6 hours either:

a. Be in at least HOT STANDBY, or

b. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_r^T to witin the limits of Figure 3.2-3 and withdraw the full length CEAs to or beyond the Long Term Steady State Insertion Limits of Specification 3.1.3.6. The THERMAL POWER limit determined from Figure 3.2-3 shall then be used to establish a revised upper THERMAL POWER level limit on Figure 3.2-4 (truncate Figure 3.2-4 at the allowable fraction of RATED THERMAL POWER determined by Figure 3.2-3) and subsequent operation shall be maintained within the reduced acceptable operation region of Figure 3.2-4.

SURVEILLANCE REQUIREMENTS

4.2.3.1 The provisions of Specification 4.0.4 are not applicable.

4.2.3.2 F_r^T shall be calculated by the expression $F_r^T = F_r(1+T_q)$ when in non-LOAD FOLLOW OPERATION and by the expression $F_r^T = 1.02\ F_r(1 + T_q)$ when in LOAD FOLLOW OPERATION. F_r^T shall be determined to be within its limit at the following intervals.

a. Prior to operation above 70 percent of RATED THERMAL POWER after each fuel loading,

b. At least once per 31 days of accumulated operation in MODE 1, and

c. Within four hours if the AXIMUTHAL POWER TILT (T_q) is > 0.03.

*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.3.3 F_r shall be determined each time a calculation of F_r^T is required by using the incore detectors to obtain a power distribution map with all full length CEAs at or above the Long Term Steady State Insertion Limit for the existing Reactor Coolant Pump combination.

4.2.3.4 T_q shall be determined each time a calculation of F_r^T is required and the value of T_q used to determine F_r^T shall be the measured value of T_q.

AZIMUTHAL POWER TILT - T_q

LIMITING CONDITION FOR OPERATION

3.2.4 The AZIMUTHAL POWER TILT (T_q) shall not exceed 0.03.

APPLICABILITY: MODE 1*

ACTION:

a. With the indicated AZIMUTHAL POWER TILT determined to be > .030 but ≤ 0.10, either correct the power tilt within two hours or determine within the next 2 hours and at least once per subsequent 8 hours, that the TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) and the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) are within the limits of Specifications 3.2.2 and 3.2.3.

b. With the indicated AZIMUTHAL POWER TILT determined to be > 0.10, operation may proceed for up to 2 hours provided that the TOTAL INTEGRATED RADIAL PEAKING FACTOR (F^T_r) and TOTAL PLANAR RADIAL PEAKING FACTOR (F^T_{xy}) are within the limits of Specifications 3.2.2 and 3.2.3. Subsequent operation for the purpose of measurement and to identify the cause of the tilt is allowable provided the THERMAL POWER level is restricted to ≤ 20% of the maximum allowable THERMAL POWER level for the existing Reactor Coolant Pump combination.

SURVEILLANCE REQUIREMENT

4.2.4.1 The provisions of Specification 4.0.4 are not applicable.

4.2.4.2 The AZIMUTHAL POWER TILT shall be determined to be within the limit by:

a. Calculating the tilt at least once per 7 days when the Subchannel Deviation Alarm is OPERABLE,

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*See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

b. Calculating the tilt at least once per 12 hours when the Subchannel Deviation Alarm is inoperable, and

c. Using the incore detectors to determine the AZIMUTHAL POWER TILT at least once per 12 hours when one excore channel is inoperable and THERMAL POWER is > 75% of RATED THERMAL POWER.

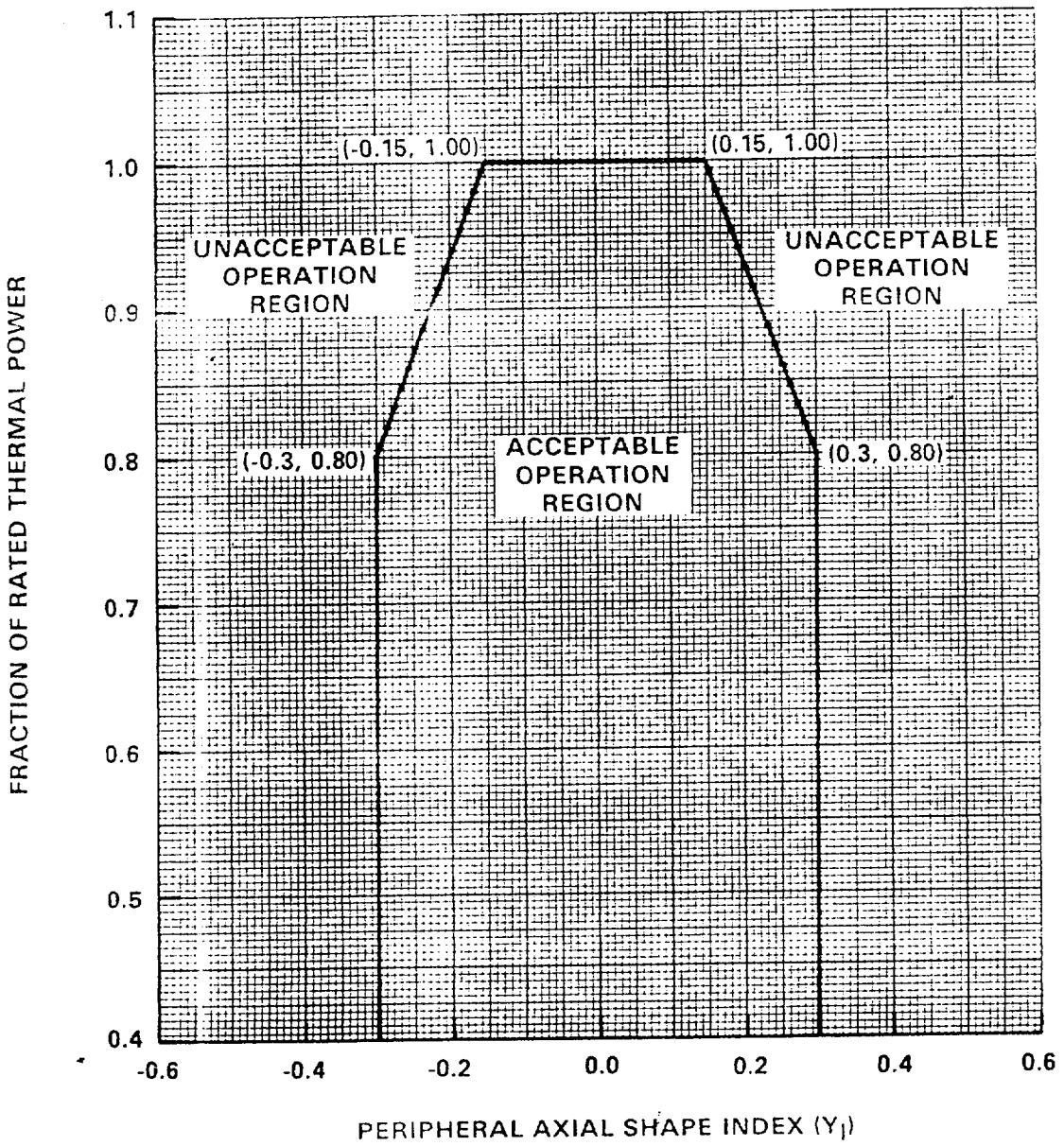

FIGURE 3.2-4
AXIAL SHAPE INDEX Operating Limits With 4 Reactor Coolant Pumps Operating

3/4.1.2 BORATION SYSTEMS (Continued)

The boron addition capability after the plant has been placed in MODES 5 and 6 requires either 1660 gallons of 8% boric acid solution from the boric acid tanks or 1630 gallons of 1720 ppm borated water from the refueling water tank to makeup for contraction of the primary coolant that could occur if the temperature is lowered from 200°F to 140°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) the potential effects of a CEA ejection accident are limited to acceptable levels.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met.

The ACTION statements applicable to an immovable or untrippable CEA and to a large misalignment ($\geq$ 15 inches) of two or more CEAs, require a prompt shutdown of the reactor since either of these conditions may be indicative of a possible loss of mechanical functional capability of the CEAs and in the event of a stuck or untrippable CEA, the loss of SHUTDOWN MARGIN.

For small misalignments (< 15 inches) of the CEAs, there is 1) a small degradation in the peaking factors relative to those assumed in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 2) a small effect on the time dependent long term power distributions relative to those used in generating LCOs and LSSS setpoints for DNBR and linear heat rate, 3) a small effect on the available SHUTDOWN MARGIN, and 4) a small effect on the ejected CEA worth used in the safety analysis. Therefore, the ACTION statement associated with the small misalignment of a CEA permits a one hour time interval during which attempts may be made to restore the CEA to within its alignment requirements prior to initiating a reduction in THERMAL POWER. The one hour time limit is sufficient to (1) identify causes of a misaligned CEA, (2) take appropriate corrective action to realign the CEAs and (3) minimize the effects of xenon redistribution.

Overpower margin is provided to protect the core in the event of a large misalignment ($\geq$ 15 inches) of a CEA. However, this misalignment would cause distortion of the core power distribution. The reactor

3/4.1.3 MOVABLE CONTROL ASSEMBLIES (Continued)

protective system would not detect the degradation in radial peaking factors and since variations in other system parameters (e.g., pressure and coolant temperature) may not be sufficient to cause trips, it is possible that the reactor could be operating with process variables less conservative than those assumed in generating LCO and LSSS setpoints. Therefore, the ACTION statement associated with the large misalignment of a CEA requires a prompt and significant reduction in THERMAL POWER prior to attempting realignment of the misaligned CEA.

The ACTION statements applicable to misaligned or inoperable CEAs include requirements to align the OPERABLE CEAs in a given group with the inoperable CEA. Conformance with these alignment requirements bring the core, within a short period of time, to a configuration consistent with that assumed in generating LCO and LSSS setpoints. However, extended operation with CEAs significantly inserted in the core may lead to perturbations in 1) local burnup, 2) peaking factors and 3) available shutdown margin which are more adverse than the conditions assumed to exist in the safety analyses and LCO and LSSS setpoints determination. Therefore, time limits have been imposed on operation with inoperable CEAs to preclude such adverse conditions from developing.

Operability of the CEA position indicators (Specification 3.1.3.3) is required to determine CEA positions and thereby ensure compliance with the CEA alignment and insertion limits and ensures proper operation of the rod block circuit. The CEA "Full In" and "Full Out" limits provide an additional independent means for determining the CEA positions when the CEAs are at either their fully inserted or fully withdrawn positions. Therefore, the ACTION statements applicable to inoperable CEA position indicators permit continued operations when the positions of CEAs with inoperable position indicators can be verified by the "Full In" or "Full Out" limits.

CEA positions and OPERABILITY of the CEA position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

The maximum CEA drop time permitted by Specification 3.1.3.4 is the assumed CEA drop time of 3.1 seconds used in the safety analyses. Measurement with $T_{avg} \geq 515°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

3/4.2.1 LINEAR HEAT RATE

The limitation on linear heat rate ensures that in the event of a LOCA, the peak temperature of the fuel cladding will not exceed 2200°F.

Either of the two core power distribution monitoring systems, the Excore Detector Monitoring System and the Incore Detector Monitoring System, provide adequate monitoring of the core power distribution and are capable of verifying that the linear heat rate does not exceed its limits. The Excore Detector Monitoring System performs this function by continuously monitoring the AXIAL SHAPE INDEX with the OPERABLE quadrant symmetric excore neutron flux detectors and verifying that the AXIAL SHAPE INDEX is maintained within the allowable limits of Figure 3.2-2. In conjunction with the use of the excore monitoring system and in establishing the AXIAL SHAPE INDEX limits, the following assumptions are made: 1) the CEA insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are satisfied, 2) the flux peaking augmentation factors are as shown in Figure 4.2-1, 3) the AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 4) the TOTAL PLANAR RADIAL PEAKING FACTOR does not exceed the limits of Specification 3.2.2.

The Incore Detector Monitoring System continuously provides a direct measure of the peaking factors and the alarms which have been established for the individual incore detector segments ensure that the peak linear heat rates will be maintained within the allowable limits of Figure 3.2-1. The setpoints for these alarms include allowances, set in the conservative directions, for 1) flux peaking augmentation factors as shown in Figure 4.2-1, 2) a measurement-calculational uncertainty factor of 1.07,* 3) an engineering uncertainty factor of 1.03, 4) an allowance of 1.01 for axial fuel densification and thermal expansion, and 5) a THERMAL POWER measurement uncertainty factor of 1.02.

3/4.2.2, 3/4.2.3 and 3/4.2.4 TOTAL PLANAR AND INTEGRATED RADIAL PEAKING FACTORS - F_{xy}^T AND F_r^T AND AZIMUTHAL POWER TILT - T_q

The limitations on F_{xy}^T and T_q are provided to ensure that the assumptions used in the analysis for establishing the Linear Heat Rate and Local Power Density - High LCOs and LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. The limitations on F_r^T and T_q are provided to ensure that the assumptions

*An uncertainty factor of 1.10 applies when in LOAD FOLLOW OPERATION.

used in the analysis establishing the DNB Margin LCO, and Thermal Margin/Low Pressure LSSS setpoints remain valid during operation at the various allowable CEA group insertion limits. If F_{xy}^T, F_r^T or T_q exceed their basic limitations, operation may continue under the additional restrictions imposed by the ACTION statements since these additional restrictions provide adequate provisions to assure that the assumptions used in establishing the Linear Heat Rate, Thermal Margin/Low Pressure and Local Power Density - High LCOs and LSSS setpoints remain valid. An AZIMUTHAL POWER TILT > 0.10 is not expected and if it should occur, subsequent operation would be restricted to only those operations required to identify the cause of this unexpected tilt.

The value of T_q that must be used in the equation $F_{xy}^T = F_{xy}\ (1 + T_q)$ and $F_r^T = F_r\ (1+T_q)$ is the measured tilt.

The surveillance requirements for verifying that F_{xy}^T, F_r^T and T_q are within their limits provide assurance that the actual values of F_{xy}^T, F_r^T and T_q do not exceed the assumed values. Verifying F_{xy}^T and F_r^T after each fuel loading prior to exceeding 75% of RATED THERMAL POWER provides additional assurance that the core was properly loaded.

3/4.2.5 DNB PARAMETERS

The limits on the DNB related parameters assure that each of the parameters are maintained within the normal steady state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the safety analyses assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR of 1.30 throughout each analyzed transient.

The 12 hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation. The 18 month periodic measurement of the RCS total flow rate is adequate to detect flow degradation and ensure correlation of the flow indication channels with measured flow such that the indicated percent flow will provide sufficient verification of flow rate on a 12 hour basis.

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

1.0 INTRODUCTION

By letter dated February 22, 1979, as supplemented April 30 and May 1, 10, 18, 22, 23 and 25, 1979, Florida Power and Light Company (FPL or the licensee) requested an amendment to the St. Lucie Plant, Unit No. 1, Facility Operating License No. DPR-67 to accommodate Cycle 3 power operation. The principal changes from Cycle 2 operations and the Cycle 2 safety analysis are the following:

- The replacement of 68 spent fuel assemblies by fresh batch E and E* assemblies.
- An increase in cycle length from approximately 8600 MD/MTU in Cycle 2 to approximately 10,000 MWD/MTU in Cycle 3.
- Technical Specification (TS) changes required to accommodate the Cycle 3 core.
- Reanalysis of Design Basis Events (DBEs) to justify operation of the Cycle 3 core.

The following analytical methodologies were modified for the Cycle 3 core analysis:

- Thermal-Hydraulic Credit for Statistical Combination previously credited to power distribution uncertainty factors is now applied to increase the allowed limit on Radial Peaking Factor (Fr).

- Peaking Factor Uncertainties of 6% in Fr and 7% in total core peaking factor (Fq) have been justified and adopted for Cycle 3.

- The coarse mesh neutronics code, ROCS, has been incorporated in the safety analysis.

- The method for computing Augmentation Factors has been modified to eliminate the over-prediction of power spiking inherent in the standard Combustion Engineering (CE) Augmentation Factor Model.

- The emergency core cooling system (ECCS) evaluation methodology has been modified to postulate two, rather than one, operable containment spray pumps.

In order to justify Cycle 3 operation the following special inspections were performed during the refueling outage:

- Inspection of CEA guide tube sleeves.

- Inspection of two CEAs which became stuck during Cycle 2 operation.

The CEA guide tube sleeving which was begun in Cycle 2 is being continued in Cycle 3.

2.0 BACKGROUND

St. Lucie Unit 1 ended Cycle 2 operation on April 1, 1979 with a total Cycle 2 burnup of 8385 MWD/MTU. St. Lucie Unit 1 operated through Cycle 2 with fuel batches A, B, C, and D at a licensed core power of 2560 MWD/MTU. Operation of Cycle 2 was at or near licensed power.

3.0 CYCLE 3 RELOAD

3.1 Core Design

The primary change to the core for Cycle 3 is the removal of the remaining 9 Batch A assemblies and 59 of the Batch B assemblies. These will be replaced by 40 Batch E assemblies and 28 Batch E* assemblies. The analysis for the Cycle 3 core was performed assuming a range of Cycle 2 burnups from 8000 to 8600 MWD/MTU to allow for flexibility in the Cycle 2 termination date. The actual Cycle 2 burnup of 8385 MWD/MTU is well within this range.

3.2 Fuel Design

3.2.1 Mechnical Design

The fresh batch E fuel is of essentially the same mechanical design as the batch D fuel introduced in Cycle 2 which we approved. Fuel assemblies of this design have had successful operating experience at Calvert Cliffs I, Fort Calhoun I, Millstone II and Maine Yankee.

The clad creep-collapse times for all fuel shown in the following table were computed using the CEPAN code (Reference T24).

Batch	Burnup for Clad Collapse (EFPH)*	Maximum Anticipated Burnup During Cycle (EFPH)*
B	≥ 24,600	23,690
C	≥ 24,600	23,690
D	≥ 24,200	14,170
E	≥ 26,900	7, 715

* 1 EFPH = 1.2830 MWD/MTU.

On the basis of the above information we find the mechanical design of the fuel acceptable.

3.2.2 Thermal Design

Using the FATES model (Reference T18), the thermal performance of the various fuel assemblies was evaluated, and the Batch E fuel was found to be limiting with respect to stored thermal energy. This is pertinent to the loss of coolant accident (LOCA) analysis of Section 9.1.

3.2.3 Chemical Design

The metallurgical requirements of the fuel cladding and the fuel assembly structural members for Batch E fuel have not been changed from the original Cycle 1 and Cycle 2 designs which have performed satisfactorily. The chemical or metallurgical performance of the Batch E fuel will be unchanged from that of the original core fuel and the discussions in the final safety analysis report (FSAR) are still valid. On this basis we find the chemical design of the Batch E fuel acceptable

3.2.4 Guide Tube Sleeving

Prior to the commencement of Cycle 2 those fuel assemblies which sustained substantial CEA guide tube wear in Cycle 1 had stainless steel sleeves installed in the guide tubes as a means of improving mechanical strength margins in worn areas. All Batch B, C. D and E fuel assemblies to be installed in CEA locations in Cycle 3 will also have stainless steel sleeves installed in the guide tubes in order to mitigate guide tube wear. A detailed discussion of the design of the sleeves and their effect on reactor operation is contained in Reference T27.

3.3 Nuclear Design

3.3.1 Fuel Management

The Batch E fuel is comprised of two sets of assemblies, each having a unique enrichment to minimize radial power peaking. The total Cycle 3 loading is 83.14 MTU. With this loading the Cycle 3 burnup for full power operation is expected to be between 9700 MWD/MTU and 10,100 MWD/MTU.

3.3.2 Nuclear Parameters

For the most part, the change from Cycle 2 nuclear parameters are small. The large increase in beginning of cycle, hot full power dissolved boron is due to greater U235 loading of the Cycle 3 core. This increased U235 loading is to accommodate the increased cycle length.

The values of physics parameters used in the safety analysis are, in all cases, either equal to or more conservative than the expected value or the limiting values in the TS.

3.4 Thermal Hydraulic Design

3.4.1 Thermal-Hydraulic Methodology

Steady state departure from nucleate boiling ratio (DNBR) analysis for Cycle 3 were performed using the same design codes as described in the FSAR. Appropriate adjustments were made to the inputs to these codes to reflect Cycle 3 parameters.

3.4.2 Thermal-Hydraulic Parameters

The Core Average Heat Flux, the Total Heat Transfer Area, and the Average Linear Heat Rate have changed slightly due to two effects: (1) The number of shims has decreased from 1296 in Cycle 2 to 588 in Cycle 3. The shims replace fuel pins, and hence there are a larger number of fuel pins in the Cycle 3 core. (2) Fuel densification is less severe in Cycle 3 than Cycle 2, leading to a greater fraction of the clad surface being available for heat transfer.

The pressure drop across the core and across the vessel have increased slightly because of the increased flow resistance of the control element assembly (CEA) guide tubes.

3.4.3 Thermal-Hydraulic Credit for Statistical Combination

In the DNB limit analysis, the assumed uncertainties in various measured parameters are not combined in a single equation but are factored into functional relationships as biases at various points in the analysis. This biasing of functional relationships throughout the analysis is equivalent to adding the absolute power uncertainties equivalent to the uncertainties in the various measured parameters and applying the total power uncertainty to the best estimate calculation. FPL has provided the following specific Cycle 3 uncertainties along with their equivalent power uncertainties:

Parameter	Parameter Uncertainty	Equivalent Power Uncertainty
ASI	0.06 ASIU	≥ 2.2%
Pressure	22 PSI	≥ 0.8%
Temperature	2 DEGF	≥ 0.9%
Flow	4 %	≥ 5.0%
Power (LSSS)	5 %	≥ 3.5%
Power (LCO)	2 %	≥ 1.4%

In the Cycle 3 analysis the equivalent sum of these uncertainties is 12.4 percent for the DNBR LSSS and 10.3 percent for the DNBR LCO. FPL believes that these components are statistically independent. We find this acceptable. This being the case, the proper method for combining these uncertainties is root sum square (RSS). The RSS combination yields 6.6 percent for the LSSS and 5.8 percent for the LCO, giving a net conservatism in the analysis of 5.8 percent for the LSSS and 4.5 percent for the LCO. For the Cycle 3 analysis FPL has proposed a partial credit of 3 percent for the LCO and LSSS. We find this acceptable.

In previous analyses of CE plants this same credit has been approved to offset the Fr measurement uncertainty. The FPL treatment differs from this in that the 3 percent credit has been applied instead to increase the allowed TS limit on Fr.

3.4.4 Thermal-Hydraulic (T-H) Effects of Guide Tube Sleeving

The T-H Effects of Guide Tube Sleeving were discussed by Northeast Nuclear Energy Company (NNEC) in Reference L1 and approved by us in Reference L3. The T-H effects for St. Lucie Unit 1 Cycle 3 are identical to those discussed by NNECO (Reference L2), and on the basis of our approval of the NNECO discussion, we find the T-H effects of Guide Tube Sleeving of St. Lucie Unit 1 Cycle 3 acceptable.

3.4.5 T-H Effect of Fuel Rod Bowing

The reduction in DNBR due to rod bowing is offset by a credit for low radial peaking in the critical assemblies. Fewer than 89 assemblies will exceed the NRC-determined penalty threshold burnup of 24,000 MWD/MTU (Reference T32) with a maximum burnup of 33,800 MWD/MTU. The corresponding DNBR penalty is 3.4%. The power distributions for Cycle 3 show the maximum radial peak for any of the 89 asssemblies to be at least 10% less than the maximum radial peak. Thus, the penalty is offset by the lower peaking of these assemblies and no power penalty for rod bowing is required for Cycle 3.

4.0 MODIFICATION OF ANALYTICAL METHODOLOGIES

4.1 PEAKING FACTOR UNCERTAINTY

In-core flux detector measurements are used to compute the core power distribution using the INCA code.

The INCA data reduction methodology and the technique for determining the required core and cycle specific coefficients are described in Reference T-19. Based on the review performed to date we find this portion of the CE INCA methodology acceptable for reloads.

The initial CE evaluation of peaking factor uncertainty was presented in CENPD-145 and CENPD-153 (References T19 and T20). In a meeting with CE on March 6, 1979, data was presented showing steady state measurement uncertainty of 6% in Fr and 7% in Fq to be conservative (Reference L10). In a previous meeting on October 12, 1978 CE had proposed that during load follow operations uncertainties of 9% in Fr and 10% in Fq should apply. Based on our review we find the above uncertainties acceptable for reloads.

For Cycle 3 operation the TS designate the appropriate peaking factor uncertainties as indicated above for steady state and load follow operations.

4.2 Use of ROCS COARSE MESH NEUTRONICS CODE

The Nuclear Design Model used in previous cycles has been based on PDQ (Reference T30), a neutron diffusion code which predicts power distribution

and fuel depletion with burnup. PDQ is a standard nuclear code accepted and used extensively throughout the industry. PDQ utilizies a simple low order differencing methodology, and because of this it is necessary to compute nuclear parameters at a large number of spatial points in order to achieve reasonable accuracy. In the Nuclear Design Model the core is represented by a two dimensional x-y cross section of the core with nuclear parameters being computed for each fuel pin, which corresponds to computing nuclear parameters of 196 points in each fuel assembly. This computational scheme is referred to as 2D PDQ. Three dimensional PDQ analyses are expensive, and hence are seldom performed.

Recently CE has begun performing some nuclear computations using the ROCS code (References T34 and T35) rather than PDQ. While CE has not yet submitted a ROCS topical report, FPL has supplied information on ROCS in the responses to requests for additional information on this reload (Reference L 12).

The ROCS code computes the same parameters as PDQ. By using a higher order differencing methodology than PDQ, ROCS is able to compute many parameters nearly as accurately as PDQ, having to compute nuclear parameters at only 1 or 4 points in the x-y plane for each assembly, rather than 196 as is done by PDQ. Also in PDQ the energy spectrum is divided into 4 energy ranges (called energy groups) whereas ROCS divides the energy spectrum into the equivalent

of one and one half energy groups (i.e., in the lower energy group only about half the nuclear parameters are represented). Because of the much coarser mesh in both space and energy allowed by ROCS, it is economically feasible to perform ROCS computations utilizing a three dimensional representation of the core, and in fact this is what is normally done.

For Cycle 3 the following safety parameters were computed using the ROCS code.

- Fuel Temperature Coefficients
- Moderator Temperature Coefficients
- Inverse Boron Worths
- Critical Boron Concentrations
- CEA drop distortion factors and reactivity worths
- Reactivity Scram Worths and Allowances
- Reactivity worth of regulating CEA bank
- Changes in 3-D core power distributions that result from inlet temperatures maldistributions (asymmetric steam generator transient)

None of these parameters requires the detailed knowledge of pin powers normally computed by 2D PDQ. CE states that in most cases these parameters are calculated more accurately by ROCS because of its ability to account for three dimensional effects. CE has stated that they observe guidelines to evaluate the adequacy of ROCS for computing these parameters on a case by case basis. If for any calculation the ROCS result is judged to be questionable, then the computation is done using PDQ.

Based on the above, we find the use of ROCS to be acceptable for this reload.

4.3 Modified Method for Computing Augmentation Factors

The densification of fuel pellets due to neutron exposure results in gaps in the fuel column. The gaps in a given fuel rod cause power spikes in nearby fuel rods and also power spikes above and below the gap in the fuel and containing the gap. CE has an NRC approved procedure for computing this power spiking which is described in Reference T18.

For this reload the licensee has identified an assumption in the CE methodology which causes the methodology to over-predict power spiking, and has proposed a revised model to give more realistic results. The assumption used by CE is that all fuel rods are equally susceptible to densification.

Test data shows, however, that the old low density fuel is more susceptible to densification than the fresh high density fuel. Furthermore, the high power density fuel rods, which are the only rods for which power spiking is a concern, are the fuel rods adjacent to the water holes within the fresh fuel assemblies. All these rods are in the immediate vicinity of only fresh fuel rods, the old fuel rods containing large gaps being at least two rod pitches away. The change the licensee has incorporated in his methodology is that he has postulated that all fuel rods less than two rod pitches away from the high power fuel

rods are fresh fuel rods with small gaps, and all fuel rods two or more rod pitches away are old fuel rods with large gaps.

We find this new methodology acceptable.

4.4 JUSTIFICATION FOR NOT RECOMPUTING TM/LP SETPOINT

The maximum Fr permitted by TS has increased by 3.9% from Cycle 2 to Cycle 3. Logically with an increase in Fr a recomputation of the thermal margin/low pressure (TM/LP) setpoint would be required. FPL has identified the following credits which more than offset the 3.9% increase in Fr, and hence assert that a recomputation of the TM/LP setpoint is not required.

- A partial credit of 3% was taken for the T-H statistical combination discussed in Section 3.4.3 of this SE
- There are fewer shims in the Cycle 3 core which increases the heat transfer area which, in turn, increases the DNBR margin by about 2%.
- There were some conservatisms in the Cycle 2 TM/LP setpoint computation.

We find this adequate justification for not recomputing the TM/LP setpoint.

5.0 SPECIFIC CYCLE 3 CONSIDERATIONS

5.1 STUCK CEA EVENT

Accordingly to Licensee Event Report 335-79-1 (Reference L7) on January 23, 1979, following a reactor trip performed as part of a shutdown procedure, two CEA stopped their fall into the core in the 8 inch withdrawn position. Subsequently they could not be moved by the CEA drive motors. It was necessary to remove the drive motors and manually pry the CEAs loose. After reconnecting the drive motors the CEAs were found to perform satisfactorily. The licensee states that there were no adverse consequences due to this occurrence.

The licensee states that the following examinations of these two CEAs were performed during the recent refueling outage:

(1) The 2 CEAs were withdrawn and reinserted twice with no indication of binding.

(2) The CEAs were thoroughly examined visually during the withdrawal/insertion operation. There was no indication of abnormal stress, wear, or binding.

(3) The withdrawal/insertion operation was performed with a load cell installed on each CEA. The load cell readings were normal.

(4) The top of each fuel assembly was visually examined. There was no indication of abnormal wear or binding.

(5) The top end of each guide tube sleeve was visually examined. Each sleeve was in place and there were no indications of deformation or damage.

(6) Both CEA assemblies were removed and transferred to the spent fuel storage pool for examination. Each CEA guide tube was vacuumed out their full length with vacuum filter examined for debris. No debris was found in any guide tube and no damage to fuel assemblies could be seen.

(7) The CEA's that stuck were dual CEA's and were both connected to the same drive mechanism in Cycle 2. Following the inspection in the fuel storage pool the assemblies were returned to different locations in the core and are now under CEA's which have separate single CEA drive units.

The licensee states that this examination gave no indication that the occurrence was related to guide tube wear, sleeving or loose parts. The licensee postulates that some debris may have temporarily collected in the lower dashpot section of a guide tube of one of the two CEA's affected. The guide tube necks down in the lower 12 inches and the sticking occurred at about 8 inches from the bottom. Therefore, it is possible that small amount of debris in the necked down section of a guide tube could cause CEA sticking. Since the reactivity in the last 8 inches of CEA travel is very small, the reactor could be safety shutdown even if all of the CEA's stuck at that level. Hot and cold drop time testing was performed, with all drop times well within limits and consistent with previous results. We find the licensee's inspection and corrective action to be adequate and concur that although the CEA sticking is not a result of an identifiable cause, it would not prevent safe shutdown of the reactor.

5.2 CEA GUIDE TUBE SLEEVE INSPECTION

Our evaluation of sleeving of CEA guide tubes at St Lucie Unit 1 is given in Section 7.0 of our safety evaluation of the cycle 2 reload, dated May 26, 1978 (Reference L4).

During the current refueling shutdown (cycle 3) the condition of guide tube sleeving was examined at St. Lucie Unit 1 by use of eddy current testing and TV scans. Similar eddy current tests and TV scans were accomplished at Millstone 2 and Calvert Cliffs 1 during refueling outages.

The eddy current tests of guide tube sleeving indicated that no detectable wear had occurred during the previous cycle at any one of the three units. Similarly, TV scans at the three units indicated proper sleeve seating. The eddy current tests at Calvert Cliff Unit 1, however, indicated inadequate crimp in some of the guide tube sleeves. Conversely, the eddy current tests at St. Lucie 1 indicated proper crimp in all guide tube sleeves. Pull tests on guide tube sleeves at Calvert Cliffs Unit 1 resulted in movement of sleeves in some of the guide tube sleeves which had shown inadequate crimp in eddy current tests. Pull tests were not performed at St. Lucie Unit 1 as there was no indication of an inadequacy of sleeve crimps during eddy current tests and no assemblies in Cycle 3 of the category where the problem has been detected (sleeved in the irradiated conditon).

The difference between the test results at St. Lucie Unit 1 and Millstone Unit 2 and those at Calvert Cliffs Unit 1 may be explained by the differences in installation of the sleeves.

At St. Lucie Unit 1 and Millstone Unit 2 pull tests were performed on the sleeves after the crimping step to verify the adequacy of the crimp. Following the "crimp verifiction" pull test, the sleeves were expanded into the guide tubes. At Calvert Cliffs Unit 1, however, the pull tests were not performed until after both the crimping and the expanding steps were completed. The licensee and CE have concluded that the method used at Calvert Cliffs Unit 1 added frictional resistance between the expanded sleeve and the guide tube to mask the presence of the inadequate crimp. The inadequate crimp would probably have been identified by an intermediate "crimp verification" pull test.

Therefore, based on the sleeving test evidence reported for St. Lucie Unit 1 and the method used for sleeving at St. Lucie we have determined that the sleeved CEA guide tubes are acceptable for cycle 3 operation. We have, however, requested that FPL provide an evaluation program which includes plans for inspections to determine guide tube wear experienced after the next cycle of operations with sleeved fuel assemblies.

6.0 REACTOR PROTECTION SYSTEM (RPS) ABNORMAL OPERATIONAL OCCURRENCES (AOOs)

The RPS AOOs, which are discussed in this section, are those postulated AOOs for which RPS trips other than the low RCS Flow Trip assure that the specified acceptable level design limits will not be violated. In the CE setpoint protection philosophy, it is postulated that complete protection rests on the combination of the limiting conditions for operation (LCO)s and limiting safety system settings (LSSS)s. However, for the transients discussed in this section, the LSSSs would provide a high degree of protection even if the LCOs were not observed.

The LCOs and LSSSs for the Cycle 3 TS were computed using the methods described in the CE SETPOINT METHODOLOGY REPORT (Reference T26). The transient plant responses used in the DBE reanalyses were calculated using the computer code CESEC (Reference T2). CESEC models both the primary and secondary coolant loops and accounts of the effects of all equipment responses in both loops.

The licensee stated that the need for reanalysis of a particular AOO is determined by comparison of the key parameters for that AOO to those of the last cycle for which a complete analysis was performed. If the key parameters are within the envelope of the reference cycle data, no reanalysis is required. A reanalysis might also be performed if it leads to a significant relaxation of TS restrictions. In some analyses, more conservative inputs than necessary are used in order that the analysis may be bounding for future cycles.

6.1 DNBR AND LHR CONSIDERATIONS

In the analysis of any postulated abnormal operational occurrence (AOO) it must be demonstrated that the DNBR limit would not be violated, i.e., that DNB does not occur, and that the linear heat rate (LHR) limit is not violated, i.e., that fuel centerline melt does not occur. In the analyses of accidents it is expected that the design limits would be violated, but the analysis must demonstrate that core damage is kept sufficiently small that the offsite exposure guidelines of 10 CFR 100 would not be violated.

CE has stated that most AOOs and Accidents are limited by DNBR considerations, and that generally the LHR values in the course of any event are so low that the LHR need not be considered in the analysis. The only events for which the LHR values become sufficiently high to merit consideration are the hot zero power (HZP) CEA Withdrawal, HZP Boron Dilution, CEA Drop, CEA Ejection and LOCA. We concur with CE's statement, and do not require LHR analysis for any except the five events cited.

6.2 TRANSIENT ANALYSIS INPUT TO SETPOINTS

6.2.1 TM/LP SETPOINT ADJUSTMENT

As stated in Reference T26, the computation of the TM/LP setpoint has as its starting point the computation of a TM/LP setpoint which guarantees that the DNBR does not decrease below 1.30 for steady state conditions. This steady state TM/LP setpoint cannot be used directly because during a transient the DNBR degrades for a short time after the trip setpoint is reached. To account for this, a time delay adjustment bias (TAB) is added to the steady state TM/LP setpoint so that the resultant dynamic TM/LP setpoint guarantees a trip soon enough in the course of the transient that core conditions will not degrade beyond the steady state TM/LP setpoint.

The TAB is determined by the transient for which the DNBR suffers the greatest degradation after the trip setpoint is reached. All potentially limiting transients must be considered in computing the TAB. Transients which do not require the TM/LP trip for production are excluded from consideration. With these criteria, the transients which must be considered in computing the TAB are the RPS AOOs which are initiated from full power, and steam generator tube rupture.

For a transient to be limiting in the determination of the TAB, it must produce a rapid degradation in DNBR. The rate of DNBR degradation was assessed, and it was determined the CEA Withdrawal Event and RCS Depressurization Event were more limiting than other competing events by a wide margin. The CEA

Withdrawal Event from Full Power produces the fastest temperature rise of any pertinent transient and the RCS depressurization event produces the fastest pressure decrease of any pertinent transient. Thus only these two events were considered in the computation of the TAB.

In the determination of the TAB from the analysis of those two events, the reactor trip is assumed to occur at the point of greatest rate of degradation of DNBR. For the purposes of computing the TAB this is a conservative approach. The minimum DNBR quoted in Table 6.0-1 for the CEA Withdrawal Event and RCS Depressurization Event is the minimum which would occur if the event were terminated by the TM/LP Trip. If some other trip is actuated before the TM/LP Trip, then the minimum DNBR would be higher. All transients other than these two are analyzed assuming the dynamic TM/LP setpoint determined by these two transients is in effect, and hence the minimum DNBR quoted for these transients in the reload application is appropriate.

6.3 RPS AOOs NOT REANALYZED

For the reasons explained in Section 4.4, even though Fr has increased in Cycle 3, there are credits which offset the resulting apparent DNBR degradation due to the Fr increase, and the demonstrated steady state DNBR margins have not degraded from their Reference Cycle values.

The Loss of Load Event produces a reactor trip in about 12 seconds in the FSAR analysis. In view of the fact that the CEA-DROP-TIME has increased by only 0.1 seconds, we have determined that the increase in CEA-DROP-TIME would have negligible impact on the Loss of Load Event.

We conclude that FPL has reanalyzed the correct set of RPS AOOS.

6.4 RPS AOOs REANALYZED

6.4.1 RCS Depressurization

The RCS Depressurization Event was reanalyzed because the CEA drop time has increased from its Reference Cycle Valve. As noted in Section 6.2, the purpose of performing this anlaysis is the computation of the TAB, which for the Cycle 3 RCS Depressurization Event is 30 psi. Since this is bounded by the 52 psi TAB computed for the hot full power (HFP) CEA Withdrawal event, the 52 psi TAB was used in computing the TM/LP setpoint. With the 52 psi TAB in effect, if the worst RCS Depressurization Event were terminated by the TM/LP Trip, the minimum DNBR would be approximately 1.34. We find this analysis and its predicted consequences acceptable.

6.4.2 HFP CEA Withdrawal

The CEA Withdrawal Events were reanalyzed because the CEA drop time has increased from its Reference Cycle value. As noted in Section 6.2 the purpose of performing the HFP analysis is the computation of the TAB, which for the Cycle 3 HFP CEA Withdrawal Event is 52 psi. Since this is greater than the 30 psi determined by the RCS Depressurization Event, the 52 psi TAB was used to determine the TM/LP Setpoint. We find this analysis and its resulting TAB acceptable. In Cycle 2 the limiting TAB was also 52 psi. In view of this fact and the discussion of Section 4.4 it is not necessary to adjust the TM/LP setpoint for Cycle 3 operation.

6.4.3 HZP CEA Withdrawal

The HZP CEA Withdrawal Event is terminated by the Variable High Power Trip at 25% power in the event of a rapid power escalation following criticality. The maximum power level attained is 148% of 2560 MWt, the maximum heat flux attained is 64.6% of 2560 MWt, and the minimum DNBR is 2.40.

The peak linear heat generation rate (PLHGR) in the course of the transient is approximately 24 KW/FT. CE states that for short transients (less than about 5 seconds) it is generally acceptable for the PLHGR to momentarily exceed the transient limit of 21 KW/FT so long as the average PLHGR for the transient is less than 21 KW/FT. For this event the power transient is approximately 3.5 seconds long and the average power is about 93% of full power. Thus the average PLHGR is well below even the steady state limit is 16 KW/FT, and the scoping criterion would indicate a low likelihood of fuel centerline melt.

The licensee has computed the total energy deposition in the fuel in the course of the transient and from this derived the peak fuel temperature. He has stated that fuel centerline melt is not predicted to occur.

We find this analysis and its predicted consequences to be acceptable.

7.1 AXIAL SHAPE INDEX (ASI) DNBR LCO ADJUSTMENT FOR TRANSIENT EFFECTS

The required overpower margin (ROPM) is a measure of the maximum DNBR degradation from the LCO which occurs in the course of a transient. The transients which must be considered in the computation of a limiting ROPM are those that are discussed in this section, all of which produce rapid degradation in DNBR.

In calculating the ROPM for these transients, the following input parameters are varied in cases where they have impact.

ASI

Power

Rod Insertion

Doppler Coefficient

Moderator Temperature Coefficient

Beta, Lambda, ℓ^*

RCS Pressure

Inlet Temperature

Axial Shapes Consistent With the Above Parameters

For each of a set of predetermined ASI values, the maximum ROPM for any of these transients and any combination of the other parameters is determined. An appropriate power to fuel design limit (PFDN) curve is chosen, which is a plot of ASI vs the power at which DNBR = 1.30. The limiting condition for operation for DNBR, based on the axial shape index, is determined by the appropriate combination of the power to fuel design limit and the required overpower margin.

There are certain transient effects which mitigate the DNBR degradation which are not factored into this analysis. Because of this the computer ROPM and this DNBR limiting condition for operation contain some conservatism above that normally identified.

We find this method of analysis and the resulting limiting condition acceptable.

The licensee states that for this reload the Four Pump Loss of Flow was the limiting transient for all values of the ASI. Thus the ROPM from the Four Pump Loss of Flow was used to compute the limiting condition for operation. This being the case, a Four Pump Loss of Flow Event initiated from any point will result in a minimum DNBR of 1.30, or slightly greater than 1.30 due to the conservatism cited above.

The Four Pump Loss of Flow analysis for this reload results in a minimum DNBR of 1.31, which is acceptable.

In the analysis of other AOOs considered in this section, the most conservative combination of initial conditions determined from the parametric study were used in the cases presented in the reload application. Hence the minimum DNBR given for these events is the minimum which can occur starting the event from any initial condition consistent with the limiting condition for operation.

7.2 AOOs USED TO ESTABLISH LCOs - NOT REANALYZED

The LCO sensitivity to the more important input parameters along with the actual changes in these parameters from the Reference Cycle values has been reviewed.

For the reasons explained in Section 4.4, even though Fr has increased in Cycle 3, there are credits which offset the resulting apparant DNBR degradation due to the Fr increase, and the demonstrated steady state DNBR margins have not degraded from their Reference Cycle values.

The Loss of AC Power Event described in the FSAR predicts a minimum DNBR of > 1.6. In view of the fact that the CEA-DROP-TIME has increased by only 0.1 seconds, we have determined that the increase in CEA-DROP-TIME would have negligible impact on the Loss of AC Power Event.

Based on Table 7.2-1 and the above arguements, we conclude that FPL has reanalyzed the correct set of AOOs.

7.3.1 Four Pump Loss of Flow

The Four Pump Loss of Flow Event was reanalyzed because the CEA Drop Time, the Radial Power Peaking, and the Axial Power Peaking have increased from their Reference Cycle values. This analysis was used to determine the LCO as described in Section 7.1. This event results in a minimum DNBR of 1.31. We find this analysis and its resulting LCO acceptable.

8.0 POSTULATED ACCIDENTS OTHER THAN LOCA

In postulated accidents a certain amount of plant and/or core damage is expected. As noted in Section 6.1 the purpose of the accident analysis is to demonstrate that in the course of an accident the offsite exposure guidelines of 10 CFR 100 are not exceeded.

8.1 ACCIDENTS NOT REANALYZED

We have reviewed the safety parameter inputs to the FPL's accident analysis and conclude that FPL has selected the correct set of accidents to reanalyze.

8.2 ACCIDENTS REANALYZED

8.2.1 CEA EJECTION

The CEA Ejection Events were reanalyzed because the CEA Drop Time and the Azimuthal Power Tilt Allowance have increased from the Reference Cycle values. Also the Cycle 3 pin census is more adverse.

The CEA Ejection Event is very rapid, and the total energy deposition is the proper parameter to monitor, rather than LHR or DNBR as is the case for other transients. The specific criteria employed are:

- Clad Damage Threshold
  Total Average Enthalpy = 200 calories/gram.
- Incipient Centerline Melting Threshold
  Total Centerline Enthalpy = 250 calories/gram.
- Fully Molten Centerline Threshold
  Total Centerline Enthalpy = 310 calories/gram.

Both the HFP and HZP cases were analyzed, and it was found that the HFP case was more limiting. For this case the analysis performed by the licensee predicted a maximum total centerline enthalpy of 289.0 calories/gram with

2.8% of the pins suffering incipent centerline melting. In the Reference Cycle the maximum total centerline enthalpy was 293.0 calories/gram, but because of the less adverse pin census in this case the fraction of fuel suffering incipent centerline melting was only 1.3%. The predicted total radial average enthalpy is less than 200 calories/gram, and hence the licensee's analysis assumes no clad damage and no radioactive release from fuel pins suffering incipient centerline melting. We find this analysis and its predicted consequences to be acceptable.

8.2.2 Siezed Rotor

The Siezed Rotor Event was reanalyzed because the radial peaking factors have increased and the pin census has become more adverse than the Reference Cycle values. In the Siezed Rotor Event, it is normal to predict a small fraction of the fuel will suffer DNB, and hence fail, and the purpose of the analysis is to assure that the radioactive release which would result from this accident is acceptable. The Cycle 3 analysis of the Siezed Rotor event predicts 0.99% cladding failure through DNB.

While we have no analysis for the radioactive release for the Siezed Rotor Event, the release can be compared to release from the CEA Ejection Event for which we do have an analysis which is acceptable. The CEA Ejection Event was previously analyzed assuming 10% failed cladding and assuming that

all radioactive release was via the path from primary coolant to secondary coolant to atmosphere, which is the same as the release path for the Siezed Rotor Event. Thus the radioactive releases for the Siezed Rotor Event with 0.99% failed fuel should be about one tenth that calculated for the CEA Ejection Event. Therefore, radioactive releases for the Siezed Rotor Event are within the 10 CFR 100 guidelines by a large margin. We find this analysis and its predicted consequences acceptable.

9.0 LOCA ANALYSIS

9.1 LARGE BREAK LOCA

The cycle 2 results for the Large Break LOCA were as follows (Reference 1):

	PCT (DEGF)	Cladding Oxidation (%)	Clad Rupture Time
Accepted Limit	2200	17	
Low Density Fuel	1952	7.63	Blowdown
High Density Fuel	1975	10.4	Reflood

The more pertinent LOCA input parameters for Cycle 2 and Cycle 3 are as follows:

Parameter	Units	Value for Limiting Fuel	
		Cycle 2 Batch D	Cycle 3 Batch E
Average LHR	KW/FT	6.2126	6.0956
Gap Conductance of PLHGR	$\frac{BTU}{HR\text{-}FT^2\text{-}DEGF}$	1552	1525
Fuel Centerline Temp at PLHGR	DEGF	3484	3512
Fuel Average Temp at PLHGR	DEGF	2184	2197
Hot Rod Gas Pressure	PSIA	1047.8	1031
Hot Rod Burnup	MWD/MTU	820	820

Although the stored energy for the limiting fuel for Cycle 3 is slightly higher than the limiting fuel for Cycle 2, the licensee states that the ECCS performance for Cycle 3 remains bounded by the previous cycle for the following reasons:

1. The cycle 2 analysis did not utilize the PARCH code (Reference T16) to compute steam cooling heat transfer coefficients after the rod rupture is predicted to occur. Since the peak clad temperature is achieved during late reflood, use of the PARCH code would have significantly reduced the peak temperatures calculated in the Cycle 2 analysis. The licensee states that use of the PARCH methodology should also reduce the calculated peak clad temperature by at least 100 DEGF for Cycle 3.

2. The radial power distribution for use in the thermal rod-to-rod radiation model is less limiting for cycle 3 than that used for the previous cycle. Use of the less limiting radiation enclosure for cycle 3 should reduce the peak clad temperatures at least 50 DEGF.

The above defined margins are far in excess of the 13 DEGF increase in fuel stored energy for the Cycle 3 limiting fuel relative to the previous cycle. The licensee concludes, therefore, that the cycle 2 analysis is also applicable to Cycle 3. In addition, we have revised the proposed TS to reduce the allowable peak LHR (Figure 3.2-1 in TS) from 14.8 KW/FT to 14.68 KW/FT. Also, we have revised Figure 3.2-2 (Axial Shape Index) to reduce the fraction of maximum allowable power level from 0.84 to 0.825. FPL has agreed to these revisions. Our calculations show that this will reduce the fuel average temperature at peak LHR from 2197 F to 2184 F which would be the same as Cycle 2. Therefore, this assures that the Cycle 3 operation is bounded by Cycle 2 analysis and is in full conformance with 10 CFR Part 50 Appendix K.

9.2 SMALL BREAK LOCA ANALYSES

On January 4, 1974 the Atomic Energy Commission issued new Acceptance Criteria for the ECCS for Light-Water Cooled Reactors (Reference T33). Subsequently CENPD-137 (Reference T15) was submitted to the AEC which demonstrated that for Calvert Cliffs class plants (2560 MWt), breaks smaller than 0.5 FT2 are not limiting. We still find this analysis acceptable, and hence the Small Break LOCA requires no reanalysis for Cycle 3.

9.3 CE RUPTURE STRAIN MODEL

The Rupture-Strain curve is a plot of Clad Strain (Clad Swelling) vs Clad Temperature at the point of clad rupture in a LOCA Event. The Rupture-Strain curve is an integral part of the CE ECCS flow blockage model. Recently the NRC staff has determined that, for clad rupture which occur during the reflood phase of the LOCA, the Rupture-Strain curve used by CE is nonconservative. That is, the CE cladding rupture-strain curve used in the flow blockage model underestimates the degree of cladding swelling in a certain portion of the curve. As a result, we informed CE of this and asked them to recompute the effect of using a rupture strain model that was conservative relative to specified portions of the relevant data and that would be acceptable under Appendix K. CE then performed some reanalysis with this curve that is conservative for System 80 plants, and they also included an improved flow distribution/heat transfer model in the vicinity of the blockage. CE has performed a calculation with these two modifications for a worst case break for System 80 plants.

We are still reviewing the new flow blockage model. We plan to complete this review by June 1979. However, we have sufficiently reviewed the information to conclude that the rupture-strain model proposed by CE is acceptable for applications to System 80 plants. Based on our review to date the flow distribution/heat transfer model used in the calculation for the System 80 worst case break is acceptable with respect to Appendix K.

System 80 plants predict clad rupture during the reflood phase of the LOCA as does St. Lucie 1. Accordingly, it is our judgment that calculations for St. Lucie 1 made with either the current model or using the revised rupture-strain curve and revised crossflow and heat transfer model, would yield comparable results. Therefore, we conclude that the St. Lucie 1 ECCS performance has been shown to conform to the applicable requirements of Appendix K to 10 CFR 50.

9.4 CONTAINMENT SPRAY ECCS ERROR

The previous ECCS analysis (Reference L1) postulated only one Containment Spray Pump operable during the reflood phase of a LOCA. Further investigation revealed that a more conservative and realistic result would be obtained if it were postulated that both Containment Spray Pumps are operable.

During the reflood the whole RCS is at containment pressure plus or minus whatever pressure is exerted by the heights of the columns of water in the RCS. The sole driving force reflooding the core is the height of water in the vessel downcomer, and the principal resistance to reflood is steam binding in the SGs. If the usual cold leg break is assumed the only path by which steam can exit the core is via the SGs. The mass flow through the SGs is proportional to the square root of the steam density, and being at essentially the same pressure as the containment, the steam density is inversely proportional to the containment pressure. Thus the lower contaiment pressure which results from having two containment spray pumps operable decreases the steam mass flow and hence decreases the reflood rate.

The previous ECCS analysis was performed based on heat transfer coefficients corresponding to the very conservative PLHGR of 17 KW/FT. The new analysis, which is performed assuming both Containment Spray Pumps are operable, assumed the TS PLHGR of 14.8 KW/FT. The new analysis is only a small segment of the total analysis intended to demonstrate the sensitivity to changing these two parameters. This new analysis predicted a PCT of 2022 DEGF, as compared with 2035 DEGF for the old analysis. Thus the combination of the two more realistic assumptions predicts a drop of 13 DEGF in PCT. We find this method for demonstrating that the original analysis is bounding to be acceptable.

10.0 TS CHANGES

The proposed Cycle 3 TS changes and the rationale for these changes are presented below.

Page 1-6 Section 1.30	The definition of LOAD FOLLOW OPERATION is introduced. This is in keeping with a change approved by us in other recent CE reload applications. The purpose of this definition is to provide a basis for determining when the higher uncertainty of TS 4.2.1.4.b.2 shall apply.
Page 2-4 Table 2.2-1	The Containment Pressure-High Trip Setpoint is changed from 3.9 psig to 3.3 psig. This is to correct a typographical error. As indicated in Reference L6, the analysis performed predicted a trip value of 3.3 psig and 3.3 psig appears in the FSAR, but 3.9 psig was inadvertently typed in the TS for Cycle 2.
Page 2-4 Table 2.2-1 Page B2-5 Section 2.2	The units of the Steam Generator Pressure-Low Trip Setpoint are changed from psig to psia. Thus, the value is changed from 485 psig to 500 psia.

Page 2-7 Figure 2.2-2

Page 3/4 2-4 Figure 3.3-2

Page 3/4 2-8a Figure 3.2-3a

Page 3/4 2-8b Figure 3.2-3b

Page 3/4 2-15 Figure 3.2-4

These figures are the LHR ASI LSSS, the LHR LCO, and the DNBR LCO. These were recomputed using Cycle 3 parameters. The LHR ASI LSSS was computed using the methods described in the CE SETPOINT METHODOLOGY Topical Report, Reference T26. In the treatment described in the CE SETPOINT METHODOLOGY REPORT, the two LCOs are maximum allowed power values given as a function of ASI. The current CE methodology is a generalization of this concept in which the maximum allowed power is given as a function of both ASI and Radial Power Peaking. We understand this concept and find the current methodology acceptable for this reload. For Cycle 3, the tents of Figures 2.2-2, 3.2-2 and 3.2-4 are more limiting than the corresponding Cycle 2 tents to compensate for the increased allowed Radial Power Peaking and the increased Azimuthal Power Tilt Allowance.

Page 3/4 1-26 Section 3.1.3.4

Page B3/4 1-4 Section 3/4.1.3

The allowed CEA drop time to 90% insertion has been increased from 3.0 seconds to 3.1 seconds. This was done in order to increase conservatism and to bound future cycle increases in CEA drop time. FPL has stated that in the most recent surveillance the longest measured CEA drop time was less than 3.0 seconds.

Page 3/4 2-1 Section 4.2.1.3	For monitoring LHR via the Excore Detectors, a requirement has been added that the CEAs shall be withdrawn at least to the Long Term Steady State Insertion Limit. This requirement is added because the LHR ASI LCO is derived assuming that the CEAs are not inserted beyond the Long Term Insertion Limit, and the LCO is not valid for the case where the CEAs are inserted farther.
Page 3/4 2-2 Section 4.2.1.3.c.2	The reference to Figure 3.2-3 is changed to Figure 3.2-3a to reflect the renumbering of TS figures.
Page 3/4 2-6 Section 3.2.2	
Page 3/4 2-2 Section 4.2.1.4.b.2	The Measured Power Distribution Uncertainty is changed to reflect the discussion of Section 4.1 of this SE.
Page B3/4 2-1 Section 3/4 2-1	
Page 3/4 2-3 Figure 3.2-1	This figure is changed to remove cycle specific details in order to extend its applicability beyond Cycle 2.
Page 3/4 2-5 Figure 4.2-1	This is a figure of Augmentation Factors vs Distance from Bottom of Core. It was computed using specific Cycle 3 parameters using the methods described in Reference T18 modified as indicated in Section 4.3 of this SE.

Page 3/4 2-6 Section 3.2.2	F_{xy}^T is increased from 1.589 to 1.627. This is consistent with the revised Figure 3.2-3a. Since this increase is properly incorporated in the Safety Analysis, we find this change acceptable.
Page 3/4 2-6 Section 4.2.2.2	In the load follow mode the measured value of F_{xy}^T is increased by 3% to account for the additional uncertainty indicated in Section 4.1 of this SE.
Page 3/4 2-6 Section 4.2.2.2 Page 3/4 2-9 Section 4.2.3.2 Page 3/4 2-11 Section 3.2.4	The Azimuthal Power Tilt Allowance has been increased from 2% to 3%. Since this increase is properly incorporated in the Safety Analysis we find this change acceptable.
Page 3/4 2-9 Section 3.2.3	The measured value of F_r^T has been increased from 1.563 to 1.64. This is consistent with the revised Figure 3.2-3b. Since this increase is properly incorporated in the Safety Analysis we find this change acceptable. Also in this section the reference to Figure 3.2-3 is changed to Figure 3.2-3b to reflect the renumbering of figures.

Page 3/4 2-9 Section 4.2.3.2	For the load follow mode the measured value of F_r^T is increased by 2% to account for the additional uncertainty indicated in Section 4.1 of this SE.

We have reviewed all TS changes and find them acceptable.

11.0 STARTUP PHYSICS TESTING

We have reviewed the Physics Startup Test Program as described in Reference L14. This program includes:

1. Rod Drop Times.
2. CEA Criticality Measurements.
3. CEA Symmetry Checks.
4. Unrodded Critical Boron Concentration Measurements.
5. Moderator Temperature Coefficient Measurements.
6. Rod Worth Measurements.
7. Shutdown Margin Verification.
8. Power Distribution Measurements at 30%, 50% and 100% Power.

Additional information as to Review and Acceptance Criteria and required Remedial Action was requested. This information was submitted in Reference L19. We have reviewed the entire Physics Startup Test Program and find it to be acceptable.

12. LICENSE CONDITION D - NEUTRON SHIELDING

By letter dated November 9, 1978, FPL notified the NRC that the additional neutron sheidling for the reactor cavity had been installed as required by License Condition D of Enclosure 1 to License No. DPR-67. The design of the shielding modification was previously reviewed and found acceptable by the NRC (Reference L4). Because the shielding is now installed, the License Condition D of Enclosure 1 may be deleted.

13.0 ENVIRONMENTAL CONSIDERATION AND CONCLUSION

13.1 ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does 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 amendment involves 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 this amendment.

13.2 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probabilty or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: May 27, 1979

14.0 REFERENCES AND METHODOLOGY CROSS INDEX

14.1 LETTER REFERENCES

L1 NNECO submits MNPS-2 Cycle 2 Reload Application, Donald C. Switzer (NNECO) to George E. Lear (NRC), September 2, 1977.

L2 FPL submits Cycle 2 Reload Application, Robert E. Uhrig (FPL) to Victor Stello (NRC), March 22, 1978.

L3 NRC approves MNPS-2 Cycle 2 Reload Application, Robert W. Reid (NRC) to Donald C. Switzer, April 19, 1978.

L4 NRC approves Cycle 2 Reload Application, Robert W. Reid (NRC) to Robert E. Uhrig (FPL), May 26, 1978.

L5 CE presents analysis to defend their Rupture-Strain ECCS Model, A.E.Scherer (CE) to Denwood F. Ross (NRC), September 18, 1978.

L6 FPL submits revision to ECCS performance results of Reference L1, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), November 9, 1978.

L7 FPL proposes TS changes, Robert E. Uhrig (FPL) to Victor Stello (NRC), January 8, 1979.

L8 FPL reports stuck CEA occurrence, A.D.Schmidt (FPL) to James P. O'Reilley (NRC), February 22, 1979.

L9 FPL submits Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Victor Stello (NRC), February 22, 1979.

L10 CE provides slides on Power Measurement Uncertainties from meeting of March 6, 1979, A.E.Scherer (CE) to Paul S. Check (NRC), March 7, 1979.

L11 NNECO provides additional information on MNPS-2 Cycle 3 Reload Application, W. G. Counsil (NNECO) to Robert W. Reid (NRC), March 27, 1979.

L12 FPL provides additional information on Cycle 3 Reload Application, Robert E Uhrig (FPL) to Robert W. Reid (NRC), April 30, 1979.

L13 BGE provides additional information on CC-1 Cycle 4 Reload Application, A. E. Lundvall (BGE) to Robert W. Reid (NRC), May 7, 1979.

L14 FPL provides information on Cycle 3 Startup Testing, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 10, 1979.

L15 FPL provides additional information on Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 18, 1979.

L16 FPL provides additional information on Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 22, 1979.

L17 FPL provides additional information on Cycle 3 Reload Application, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 23, 1979.

L18 FPL provides information on CEA Guide Tube Sleeving Inspection, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 23, 1979.

L19 FPL provides information on Cycle 3 Startup Testing, Robert E. Uhrig (FPL) to Robert W. Reid (NRC), May 25, 1979.

14.2 TOPICAL REFERENCES AND REVIEW STATUS

REF. No.	REFERENCE TITLE and REVIEW STATUS
T1	CENPD-98, COAST Code Description," April 1973. Review Status: Approved.
T2	CENPD-107, "CESEC - Digital Simulation of a C-E Nuclear Steam Supply System," April 1974. Review Status: Still under review. Initial NRC comparison of CESEC computations with RELAP-3 computations show good agreement. Considered to be acceptable for reload analyses.
T3	CENPD-132, "Calculative Methods for the CE Large Break LOCA Evaluation Model," August 1974. Review Status: Approved.
T4	CENPD-132, Supplement 1, "Updated Calculative Methods for the CE Large Break LOCA Evaluation Model," December 1974. Review Status: Approved
T5	CENPD-132, Supplement 2, "Calculational Methods for the CE Large Break LOCA Evaluation Model," July 1975. Review Status: Approved.

T6 CENPD-133, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis," April 1974.
Review Status: Approved.

T7 CENPD-133, Supplement 1, "CEFLASH-4AS, A Computer Program for Reactor Blowdown Analysis of the Small Break Loss-of-coolant Accident," August 1974.
Review Status: Approved.

T8 CENPD-133, Supplement 2, "CEFLASH-4A, A FORTRAN IV Digital Computer Program for Reactor Blowdown Analysis (Modification)," December 1974.
Review Status: Approved.

T9 CENPD-134, "COMPERC-II, A Program for Emergency Refill-Reflood of the Core," April 1974.
Review Status: Approved.

T10 CENPD-134, Supplement 1, "COMPREC-II, A Program for Emergency Refill-Reflood of the Core (Modification)," December 1974.
Reveiw Status: Approved.

T11 CENPD-135-P, "STRIKIN II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1974.
Review Status: Approved.

T12 CENPD-135, Supplement 2-P, "STRINKIN- , A Cylindrical Geometry Fuel Rod Heat Transfer Program (Modification)," February 1975.
Review Status: Approved.

T13 CENPD-135, Supplement 4-P, "STRINKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," August 1976.
Review Status: Approved.

T14 CENPD-135, Supplement 5-P, "STRINKIN-II, A Cylindrical Geometry Fuel Rod Heat Transfer Program," April 1977.
Review Status: Approved.

T15 CENPD-137, "Calculative Methods for the C-E Small Break LOCA Evaluation Model," August 1974.
Review Status: Approved

T16 CENPD-138, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," August 1974.
Review Status: Approved.

T17 CENPD-138, Supplement 1, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup" (Modification), February 1975.
Review Status: Approved.

T18 CENPD-139, "CE Fuel Evaluation Model," July 1974.
Review Status: Approved.

T19 CENPD-145, "INCA: Method of Analyzing In-Core Detector Data in Power Reactors," Combustion Engineering, April 1975.
Review Status: Still under review. INCA computational methodology appears to be reasonable the uncertainty analysis in these topical's is not acceptable. Based on subsequent discussion and submittals, the interim assumed measurement uncertainties of 6% in Fr and 7% in Fq appear to be reasonable.

T20 CENPD-153,"Evaluation of Uncertainty in the Nuclear Form Factor Measured by Self-Powered Fixed In-Core Detector Systems", Combustion Engineering, August 1974.
Review Status: Same as that for T19.

T21 CENPD-161-P, "TORC Code - A Computer Code for Determining the Thermal Margin of a Reactor Core," July 1975.
Review Status: Approved.

T22 CENPD-162-P-A, "CE Critical Heat Flux," September 1976.
Review Status: Approved.

T23 CENPD-183, "CE Methods for Loss of Flow Analysis," July 1975.
Review Status: Review has progressed to the point where this methodology is considered to be acceptable for reload analyses.

T24 CENPD-187, "CEPAN Method of Analyzing Creep Collapse of Oval Cladding," June 1975.

Review Status: Approved.

T25 CENPD-190A, "CE Ejection, C-E Method for Control Element Assembly Ejection," July 1976.

Review Status: Approved.

T26 CENPD-199-P, "C-E Setpoint Methodology," April 1976.

Review Status: Review has progressed to the point where this methodology is considered to be acceptable for reload analyses.

T27 CEN-80(N)-P, "Millstone Unit 2 Reactor Operation With Modified CEA Guide Tubes," February 8, 1978.

T28 CEN-89(F)-P, "Solution to Increased Water Hole Peaking in Operating Reactors St. Lucie 1, P. C. Uhrig to R. Reid, April 10, 1978.

Review Status: New TM-LP methodology appears reasonable. Uncertainty analysis presented is not acceptable, but has been superceded by subsequent analysis by CE, as indicated for T19 and T20 above.

T29 WAPD-TM-479, J. A. Redfield, "CHIC-KIN - A Fortran Program for Intermediate and Fast Transients in a Water Moderator Reactor," January 1965.
Review Status: Standard Nuclear Industry Code.

T30 WAPD-TM-678, W.R. Cadwell, "PDQ-7 Reference Manual," January 1978.
Review Status: Standard Nuclear Industry Code.

T31 WAPD-TM-743, J. B. Yasinsky, M. Natelson, and L.A. Hageman, "TWIGL - A Program to Solve the Two-Dimensional, Two Group, Space-Time Neutron Diffusion Equations with Temperature Feedback," February 1968.
Review Status: Standard Nuclear Industry Code.

T32 Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors (NRC Report) Pages 21 and 26.
Review Status: Approved.

T33 Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Federal Register, Vol. 39, No. 3, January 4, 1974.
Review Status: Approved.

T34 A Higher Ordered Difference Method for Diffusion Theory, C. P. Robinson & J. D. Eckhard, Trans. Am. Nucl. Soc., 15, Page 297, 1972.

Review Status: Standard Periodical Literature.

T35 Theory, Capabilities, and Use of the Three-Dimensional Reactor Operation and Control Simulator (ROCS), T. G. Ober, J.C. Stork, I.C. Rickard & J.K. Gasper, Nuclear Sciences & Engineering, 64, pp 605-623, 1977.

Review Status: Standard Periodical Literature.

14.3 METHODOLOGY CROSS INDEX FOR DBEs

DBE	References
RCS Depressurization	T2, T21, T22, T26*, T28
CEA Withdrawal	T2, T21, T22, T26*, T28
Loss of Coolant Flow	T1, T2, T11, T21, T22, T23*, T26
Full Length CEA Drop	T2, T21, T22, T26*
CEA Ejection	T25*, T29, T30, T31
Siezed Rotor	FSAR*, T2, T11, T21, T22, T30
Large Break LOCA	FSAR*, T3, T4, T5, T6, T7, T8, T9, T10,T11, T12, T13, T14, T17, T18
Small Break LOCA	T16*

*The asterisk indicates the reference in which the DBE and its associated analytical methodology is described. In all cases the event is described in the FSAR, but in those cases where the FSAR is not indicated a more complete description of the event and its associated analytical methodology is given in the reference topical indicated by an asterisk.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 32 to Facility Operating License No. DPR-67 issued to Florida Power & Light Company (the licensee), which revised the license and its appended Technical Specifications for operation of St. Lucie Plant, Unit No. 1 (the facility), located in St. Lucie County, Florida. The amendment is effective as of its date of issuance.

This amendment revises the Technical Specifications to incorporate changes resulting from Cycle 3 reload fuel, and deletes a license requirement for neutron shielding which has been satisfactorily completed.

The applications for the amendment 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 Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

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For further details with respect to this action, see (1) the applications for amendment dated November 9, 1978 and February 22, 1979, as supplemented April 30 and May 1, 10, 18, 22 and 23, 1979, (2) Amendment No. 32 to License No. DPR-67, 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 Indian River Junior College Library, 3209 Virginia Avenue, Ft. Pierce, Florida. 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 Operating Reactors.

Dated at Bethesda, Maryland this 27th day of May 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Operating Reactors