DCS MS-016

Docket No. 50-335

MAR 1 1984

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Mr. J. W. Williams, Jr.
Vice President
Nuclear Energy Department
Florida Power & Light Company
P. O. Box 14000
Juno Beach, Florida 33408

Dear Mr. Williams:

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. DPR-67 for the St. Lucie Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated January 20, 1983 as supplemented February 8 and 17, March 11, April 6, May 13, June 23 and July 15, 1983.

The amendment revises the Technical Specifications to change the shutdown margin requirements, change the moderator temperature coefficient limits and delete the flux peaking augmentation curve for fuel Cycle 6 and adds a license condition that restricts operation to 38,000 MWd/MTU peak assembly for CE fuel unless certain conditions are met.

On the basis of our review, we conclude that the fuel design is acceptable with the following follow-up action required:

> Prior to reaching 38,000 MWd/MTU peak assembly, you must use an approved method to show that the CE fuel will not experience collapse unless the new ENC methodology has been approved by the staff.

The transient analyses for Cycle 6 entails a change in fuel vendor and analytical methodology. The PTSPWR2 computer code was used for much of the ENC transient analysis. Since this code is presently under staff review, the acceptability of the Cycle 6 transient analysis was based primarily on comparison between ENC results and previously reported results by CE. However, should the staff's review of the PTSPWR2 code warrant a need for reanalysis of transient events, you will be required to submit a modified analysis in conformance with our conclusions. In the steam line break analysis, you did not model the asymmetric thermal-hydraulic and neutronic system behavior. You must provide confirmatory re-analysis of the event prior to the next refueling. In addition, you need to provide a reassessment of the limiting steam line break event in accordance with the guidance outlined in Section 15.1.5 of the Standard Review Plan (NUREG-0800).

The loss of coolant accident (LOCA) was analyzed by ENC using the EXEM/PWR ECCS model which is still under staff review. The review has progressed to a point where we can conclude that the use of the EXEM/PWR evaluation model is acceptable for this reload. However, should the staff's review of the EXEM/PWR model warrant a need for reanalysis of the LOCA, you will be required to submit modified analysis in conformance with our conclusions. Prior to the next reload, you should also provide justification that the worst assumption regarding single failure has been considered.

A copy of the related Safety Evaluation is also enclosed. The Notice of Issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Original signed by
CMTrammell
for James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No63 to DPR-67
2. Safety Evaluation

cc w/enclosures:
See next page

CMT 3/1/84

ORB#3:DL	ORB#3:DL	ORB#3:DL	OELD	AD:OR:DL
PMKreutzer	DESells/pn	JRMiller	~~[illegible]~~ W.D. Paton	GCLainas
2/ /84	2/ /84	2/ /84	~~2/16/84~~ 2/27/84	2/30/84

Florida Power & Light Company

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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. 63
License No. DPR-67

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

   A. The application for amendment by Florida Power & Light Company, (the licensee) dated January 20, 1983 as supplemented, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, Facility Operating License No. DPR-67 is amended by changes to the Technical Specifications as indicated in the Attachment to this license amendment by amending paragraph 2.C(2), and by adding a new paragraph 2.C(4) to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   (4) Prior to reaching 38,000 MWd/MTU peak assembly, the licensee must use an approved method to show that Combustion Engineering fuel will not experience creep collapse unless the new Exxon Corporation methodology has been approved for use by the staff and its results are valid for Cycle 6.

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

FOR THE NUCLEAR REGULATORY COMMISSION

for Charles M. Trammell
James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: March 1, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 63

TO FACILITY OPERATING LICENSE NO. DPR-67

DOCKET NO. 50-335

Replace the following pages of the Appendix "A" Technical Sepcifications 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.

Pages

B 2-1
B 2-3
B 2-5
B 2-7
3/4 1-1
3/4 1-5
3/4 2-2
3/4 2-5
B 3/4 1-1
B 3/4 2-1
B 3/4 4-1

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this safety limit prevent overheating of the fuel cladding and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate below the level at which centerline fuel melting will occur. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

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

The minimum value of the DNBR during steady state operation, normal operational transients, and anticipated transients is limited to 1.22 using the XNB DNBR correlation. This value corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur and is chosen as an appropriate margin to DNB for all operating conditions.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and maximum cold leg temperature with four Reactor Coolant Pumps operating for which the minimum DNBR is no less than the DNBR limit for the family of axial shapes and corresponding radial peaks shown in Figure B 2.1-1. The limits in Figure 2.1-1 were calculated for reactor coolant inlet temperatures less than or equal to 580°F. The dashed line at 580°F coolant inlet temperature is not a safety limit; however, operation above 580°F is not possible because of the actuation of the main steam line safety valves which limit the maximum value of reactor inlet temperature. Reactor operation at THERMAL POWER levels higher than 112% of RATED THERMAL POWER is prohibited by the high power level trip setpoint specified in Table 2.1-1. The area of safe operation is below and to the left of these lines.

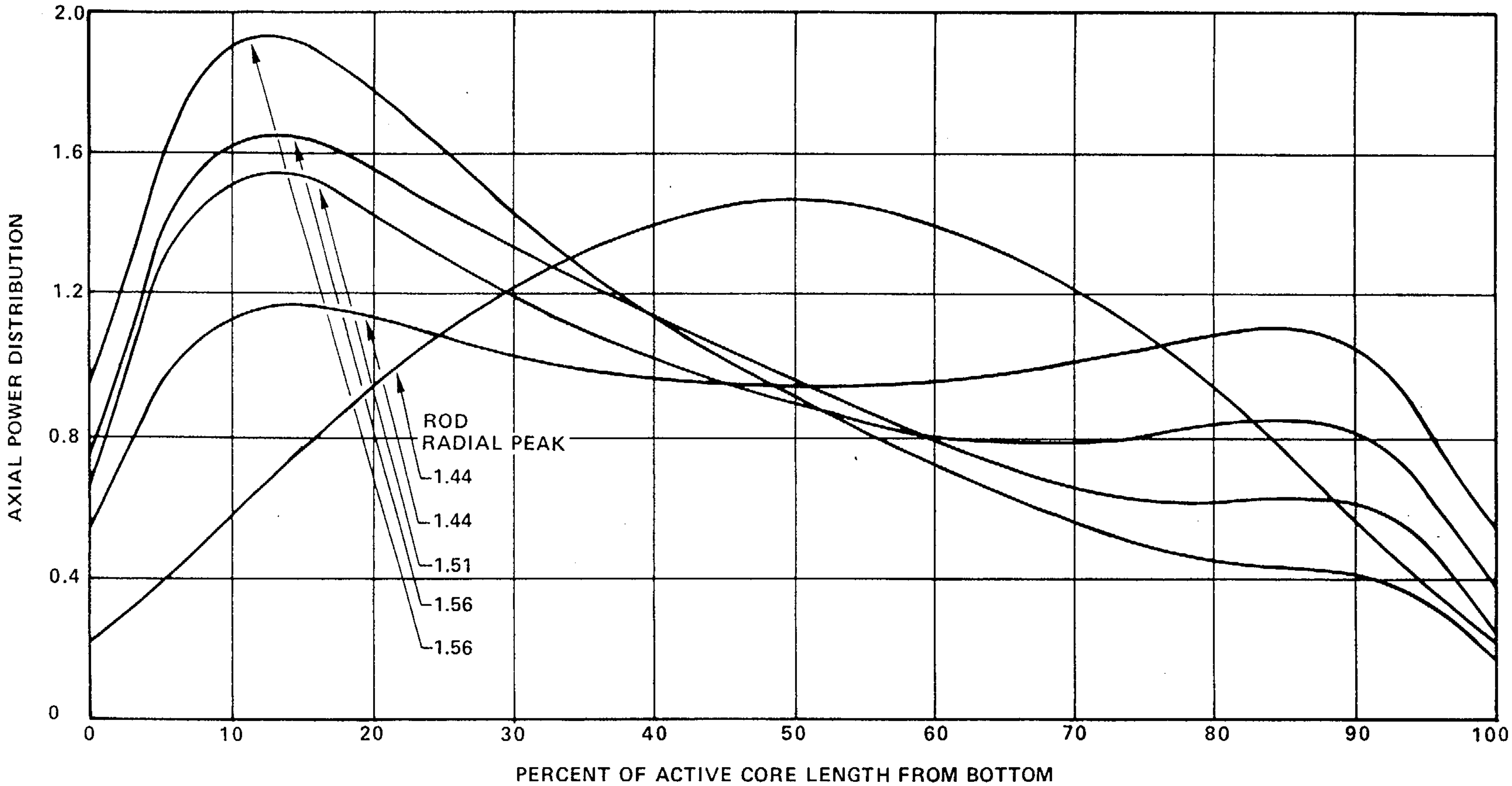

Figure B2.1-1 Axial Power Distribution for Thermal Margin Safety Limits

The conditions for the Thermal Margin Safety Limit curves in Figure 2.1-1 to be valid are shown on the figure.

The reactor protective system in combination with the Limiting Conditions for Operation is designed to prevent any anticipated combination of transient conditions for reactor coolant system temperature, pressure, and thermal power level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities.

2.1.2 REACTOR COOLANT SYSTEM PRESSURE

The restriction of this Safety Limit protects the integrity of the Reactor Coolant System from overpressurization and thereby prevents the release of radionuclides contained in the reactor coolant from reaching the containment atmosphere.

The reactor pressure vessel and pressurizer are designed to Section III of the ASME Code for Nuclear Power Plant components which permits a maximum transient pressure of 110% (2750 psia) of design pressure. The Reactor Coolant System piping, valves and fittings are designed to ANSI B 31.7, Class I which permits a maximum transient pressure of 110% (2750 psia) of component design pressure. The Safety Limit of 2750 psia is therefore consistent with the design criteria and associated code requirements.

The entire Reactor Coolant System is hydrotested at 3125 psia to demonstrate integrity prior to initial operation.

2.2 LIMITING SAFETY SYSTEM SETTINGS

BASES

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 the DNBR limit 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 the DNBR limit 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 is 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 600 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 $\pm$ 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.

Thermal Margin/Low Pressure

The Thermal Margin/Low Pressure trip is provided to prevent operation when the DNBR is less than the DNBR limit.

The trip is initiated whenever the reactor coolant system pressure signal drops below either 1887 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of reactor coolant pumps operating and the AXIAL SHAPE INDEX. The minimum value of reactor coolant flow rate, the maximum AZIMUTHAL POWER TILT and the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with 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.

The Thermal Margin/Low Pressure trip setpoints include appropriate allowances for equipment response time, calculational and measurement uncertainties, and processing error. A further allowance of 30 psia is included to compensate for the time delay associated with providing effective termination of the occurrence that exhibits the most rapid decrease in margin to the DNBR limit.

Asymmetric Steam Generator Transient Protective Trip Function (ASGTPTF)

The ASGTPTF consists of Steam Generator pressure inputs to the TM/LP calculator, which causes a reactor trip when the difference in pressure between the two steam generators exceeds the trip setpoint. The ASGTPTF is designed to provide a reactor trip for those events associated with secondary system malfunctions which result in asymmetric primary loop coolant temperatures. The most limiting event is the loss of load to one steam generator caused by a single main steam isolation valve closure.

The equipment trip setpoint and allowable values are calculated to account for instrument uncertainties, and will ensure a trip at or before reaching the analysis setpoint.

Loss of Turbine

A Loss of Turbine trip causes a direct reactor trip when operating above 15% of RATED THERMAL POWER. This trip provides turbine protection, reduces the severity of the ensuing transient and helps avoid the lifting of the main steam line safety valves during the ensuing transient, thus extending the service life of these valves. No credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

Rate of Change of Power-High

The Rate of Change of Power-High trip is provided to protect the core during startup operations and its use serves as a backup to the administratively enforced startup rate limit. Its trip setpoint does not correspond to a Safety Limit and no credit was taken in the accident analyses for operation of this trip. Its functional capability at the specified trip setting is required to enhance the overall reliability of the Reactor Protection System.

3/4.1 REACTIVITY CONTROL SYSTEMS

3/4.1.1 BORATION CONTROL

SHUTDOWN MARGIN - T_{avg} > 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.1 The SHUTDOWN MARGIN shall be $\geq$ 3.6% $\Delta k/k$.

APPLICABILITY: MODES 1, 2*, 3 and 4.

ACTION:

With the SHUTDOWN MARGIN < 3.6% $\Delta k/k$, immediately initiate and continue boration at $\geq$ 40 gpm of 1720 ppm boron or equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.1.1 The SHUTDOWN MARGIN shall be determined to be $\geq$ 3.6% $\Delta k/k$:

a. Within one hour after detection of an inoperable CEA(s) and at least once per 12 hours thereafter while the CEA(s) is inoperable. If the inoperable CEA is immovable or untrippable, the above required SHUTDOWN MARGIN shall be increased by an amount at least equal to the withdrawn worth of the immovable or untrippable CEA(s).

b. When in MODES 1 or 2#, at least once per 12 hours by verifying that CEA group withdrawal is within the Power Dependent Insertion Limits of Specification 3.1.3.6.

c. When in MODE 2##, at least once during CEA withdrawal and at least once per hour thereafter until the reactor is critical.

d. Prior to initial operation above 5% RATED THERMAL POWER after each fuel loading, by consideration of the factors of e below, with the CEA groups at the Power Dependent Insertion Limits of Specification 3.1.3.6.

*See Special Test Exception 3.10.1.

#With $K_{eff} \geq 1.0$.

##With $K_{eff} < 1.0$.

SURVEILLANCE REQUIREMENTS (Continued)

e. When in MODES 3 or 4, at least once per 24 hours by consideration of the following factors:

1. Reactor coolant system boron concentration,
2. CEA position,*
3. Reactor coolant system average temperature,
4. Fuel burnup based on gross thermal energy generation,
5. Xenon concentration, and
6. Samarium concentration.

4.1.1.1.2 The overall core reactivity balance shall be compared to predicted values to demonstrate agreement within $\pm$ 1.0% $\Delta k/k$ at least once per 31 Effective Full Power Days (EFPD). This comparison shall consider at least those factors stated in Specification 4.1.1.1.1.e, above. The predicted reactivity values shall be adjusted (normalized) to correspond to the actual core conditions prior to exceeding a fuel burnup of 60 Effective Full Power Days after each fuel loading.

*For Modes 3 and 4, during calculation of shutdown margin with all CEA's verified fully inserted, the single CEA with the highest reactivity worth need not be assumed to be stuck in the fully withdrawn position.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.4 The moderator temperature coefficient (MTC) shall be:

a. Less positive than 0.7×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is $\leq$ 70% of RATED THERMAL POWER,

b. Less positive than 0.2×10^{-4} $\Delta k/k/°F$ whenever THERMAL POWER is > 70% of RATED THERMAL POWER, and

c. Less negative than -2.8×10^{-4} $\Delta k/k/°F$ at RATED THERMAL POWER.

APPLICABILITY: MODES 1 and 2*#

ACTION:

With the moderator temperature coefficient outside any one of the above limits, be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.4.1 The MTC shall be determined to be within its limits by confirmatory measurements. MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the above limits.

*With $K_{eff} \geq 1.0$.

#See Special Test Exception 3.10.2.

SURVEILLANCE REQUIREMENTS (Continued)

4.1.1.4.2 The MTC shall be determined at the following frequencies and THERMAL POWER conditions during each fuel cycle:

a. Prior to initial operation above 5% of RATED THERMAL POWER, after each refueling.

b. At any THERMAL POWER, within 7 EFPD after initially reaching a RATED THERMAL POWER equilibrium boron concentration.

c. At any THERMAL POWER, within 7 EFPD after reaching a RATED THERMAL POWER equilibrium boron concentration of 300 ppm.

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.

SURVEILLANCE REQUIREMENTS (Continued)

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_{xy}^{T} 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. A measurement-calculational uncertainty factor of 1.07,*

2. An engineering uncertainty factor of 1.03,

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

4. A THERMAL POWER measurement uncertainty factor of 1.02.

#If the core system becomes inoperable, reduce power to M x N within 4 hours and monitor linear heat rate in accordance with Specification 4.2.1.

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

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TOTAL PLANAR RADIAL PEAKING FACTOR - F_{xy}^T

LIMITING CONDITION FOR OPERATION

3.2.2 The calculated value of F_{xy}^T, defined as $F_{xy}^T = F_{xy}(1+T_q)$, shall be limited to $\leq$ 1.70.

APPLICABILITY: MODE 1*.

ACTION:

With F_{xy}^T > 1.70, within 6 hours either:

a. Reduce THERMAL POWER to bring the combination of THERMAL POWER and F_{xy}^T 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_{xy}^T shall be calculated by the expression $F_{xy}^T = F_{xy}(1+T_q)$ when in non-LOAD FOLLOW OPERATION and by the expression $F_{xy}^T = 1.03\ F_{xy}(1+T_q)$ when in LOAD FOLLOW OPERATION. F_{xy}^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 AZIMUTHAL POWER TILT (T_q) is > 0.03.

*See Special Test Exception 3.10.2.

3/4.1.1 BORATION CONTROL

3/4.1.1.1 and 3/4.1.1.2 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

SHUTDOWN MARGIN requirements vary throughout core life as a function of fuel depletion, RCS boron concentration, and RCS T_{avg}. The most restrictive condition occurs at EOL, with T_{avg} at no load operating temperature, and is associated with a postulated steam line break accident and resulting uncontrolled RCS cooldown. In the analysis of this accident, a minimum SHUTDOWN MARGIN of 3.6% $\Delta k/k$ is required to control the reactivity transient. Accordingly, the SHUTDOWN MARGIN required by Specification 3.1.1.1 is based upon this limiting condition and is consistent with FSAR accident analysis assumptions. For earlier periods during the fuel cycle, this value is conservative. With $T_{avg} \leq 200°F$, the reactivity transient resulting from a boron dilution event with a partially drained Reactor Coolant System requires a 2% $\Delta k/k$ SHUTDOWN MARGIN and restrictions on charging pump operation to provide adequate protection. A 2% $\Delta k/k$ SHUTDOWN MARGIN is 1.0% $\Delta k/k$ conservative for Mode 5 operation with total RCS volume present, however LCO 3.1.1.2 is written conservatively for simplicity.

3/4.1.1.3 BORON DILUTION AND ADDITION

A minimum flow rate of at least 3000 GPM provides adequate mixing, prevents stratification and ensures that reactivity changes will be gradual during boron concentration changes in the Reactor Coolant System. A flow rate of at least 3000 GPM will circulate an equivalent Reactor Coolant System volume of 11,400 cubic feet in approximately 26 minutes. The reactivity change rate associated with boron concentration changes will be within the capability for operator recognition and control.

3/4.1.1.4 MODERATOR TEMPERATURE COEFFICIENT (MTC)

The limiting values assumed for the MTC used in the accident and transient analyses were $+ 0.7 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels $\leq$ 70% of RATED THERMAL POWER, $+ 0.2 \times 10^{-4}$ $\Delta k/k/°F$ for THERMAL POWER levels > 70% of RATED THERMAL and $- 2.8 \times 10^{-4}$ $\Delta k/k/°F$ at RATED THERMAL POWER. Therefore, these limiting values are included in this specification. Determination of MTC at the specified conditions ensures that the maximum positive and/or negative values of the MTC will not exceed the limiting values.

3/4.1.1.5 MINIMUM TEMPERATURE FOR CRITICALITY

The MTC is expected to be slightly negative at operating conditions. However, at the beginning of the fuel cycle, the MTC may be slightly positive at operating conditions and since it will become more positive at lower temperatures, this specification is provided to restrict reactor operation when T_{avg} is significantly below the normal operating temperature.

3/4.1.2 BORATION SYSTEMS

The boron injection system ensures that negative reactivity control is available during each mode of facility operation. The components required to perform this function include 1) borated water sources, 2) charging pumps, 3) separate flow paths, 4) boric acid pumps, 5) associated heat tracing systems, and 6) an emergency power supply from OPERABLE diesel generators.

With the RCS average temperature above 200°F, a minimum of two separate and redundant boron injection systems are provided to ensure single functional capability in the event an assumed failure renders one of the systems inoperable. Allowable out-of-service periods ensure that minor component repair or corrective action may be completed without undue risk to overall facility safety from injection system failures during the repair period.

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 2.0% $\Delta k/k$ after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 7,925 gallons of 8.0% boric acid solution from the boric acid tanks or 13,700 gallons of 1720 ppm borated water from the refueling water tank.

The requirements for a minimum contained volume of 401,800 gallons of borated water in the refueling water tank ensures the capability for borating the RCS to the desired level. The specified quantity of borated water is consistent with the ECCS requirements of Specification 3.5.4. Therefore, the larger volume of borated water is specified here too.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

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, provides adequate monitoring of the core power distribution and is 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 AZIMUTHAL POWER TILT restrictions of Specification 3.2.4 are satisfied, and 3) 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) a measurement-calculational uncertainty factor of 1.07,* 2) an engineering uncertainty factor of 1.03, 3) an allowance of 1.01 for axial fuel densification and thermal expansion, and 4) 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^T_{xy} AND F^T_r AND AZIMUTHAL POWER TILT - T_q

The limitations on F^T_{xy} 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^T_r 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^T_{xy}, F^T_r 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 $F^T_{xy} = F_{xy} (1 + T_q)$ and $F^T_r = F_r (1+T_q)$ is the measured tilt.

The surveillance requirements for verifying that F^T_{xy}, F^T_r and T_q are within their limits provide assurance that the actual values of F^T_{xy}, F^T_r and T_q do not exceed the assumed values. Verifying F^T_{xy} and F^T_r 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 $\geq$ 1.22 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.

3/4.4.1 REACTOR COOLANT LOOPS AND COOLANT CIRCULATION

The plant is designed to operate with both reactor coolant loops and associated reactor coolant pumps in operation, and maintain DNBR above the DNBR limit during all normal operations and anticipated transients. In MODES 1 and 2 with one reactor coolant loop not in operation, this specification requires that the plant be in at least HOT STANDBY within 1 hour.

In MODE 3, a single reactor coolant loop provides sufficient heat removal capability for removing decay heat; however, single failure considerations require that two loops be OPERABLE.

In MODE 4, and in MODE 5 with reactor coolant loops filled, a single reactor coolant loop or shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations require that at least two loops (either shutdown cooling or RCS) be OPERABLE.

In MODE 5 with reactor coolant loops not filled, a single shutdown cooling loop provides sufficient heat removal capability for removing decay heat; but single failure considerations and the unavailability of the steam generators as a heat removing component, require that at least two shutdown cooling loops be OPERABLE.

The operation of one Reactor Coolant Pump or one shutdown cooling pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the Reactor Coolant System. The reactivity change rate associated with boron reductions will, therefore, be within the capability of operator recognition and control.

The restrictions on starting a Reactor Coolant Pump in MODE 5 with one or more RCS cold legs less than or equal to 165°F are provided to prevent RCS pressure transients, caused by energy additions from the secondary system, which could exceed the limits of Appendix G to 10 CFR 50. The RCS will be protected against overpressure transients and will not exceed the limits of Appendix G by either 1) restricting the water volume in the pressurizer and thereby providing a volume for the primary coolant to expand into, or 2) by restricting starting of the Reactor Coolant Pumps to when the secondary water temperature of each steam generator is less than 45°F above each of the Reactor Coolant System cold leg temperatures.

3/4.4.2 and 3/4.4.3 SAFETY VALVES

The pressurizer code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2750 psia. Each safety valve is designed to relieve 2 x 10^5 lbs per hour of saturated steam at the valve setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an operating shutdown cooling loop, connected to the RCS, provides overpressure relief capability and will prevent RCS overpressurization.

3/4.4.2 and 3/4.4.3 SAFETY VALVES (Continued)

During operation, all pressurizer code safety valves must be OPERABLE to prevent the RCS from being pressurized above its safety limit of 2750 psia. The combined relief capacity of these valves is sufficient to limit the Reactor Coolant System pressure to within its Safety Limit of 2750 psia following a complete loss of turbine generator load while operating at RATED THERMAL POWER and assuming no reactor trip until the first Reactor Protective System trip setpoint (Pressurizer Pressure-High) is reached (i.e., no credit is taken for a direct reactor trip on the loss of turbine) and also assuming no operation of the pressurizer power operated relief valve or steam dump valves.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code, 1974 Edition.

3/4.4.4 PRESSURIZER

A steam bubble in the pressurizer ensures that the RCS is not a hydraulically solid system and is capable of accommodating pressure surges during operation. The steam bubble also protects the pressurizer code safety valves and power operated relief valve against water relief. The power operated relief valve and steam bubble function to relieve RCS pressure during all design transients. Operation of the power operated relief valve in conjunction with a reactor trip on a Pressurizer-Pressure-High signal minimizes the undesirable opening of the spring-loaded pressurizer code safety valves. The required pressurizer heater capacity is capable of maintaining natural circulation subcooling. Operability of the heaters, which are powered by a diesel generator bus, ensures ability to maintain pressure control even with loss of offsite power.

3/4.4.5 STEAM GENERATORS

One OPERABLE steam generator provides sufficient heat removal capability to remove decay heat after a reactor shutdown. The requirement for two steam generators capable of removing decay heat, combined with the requirements of Specifications 3.7.1.1, 3.7.1.2 and 3.7.1.3 ensures adequate decay heat removal capabilities for RCS temperatures greater than 325°F if one steam generator becomes inoperable due to single failure considerations. Below 325°F, decay heat is removed by the shutdown cooling system.

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical demage or progressive degradation due to design, manufacturing errors, or inservice conditions that lead to corrosion. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE UNIT 1

DOCKET NO. 50-335

1.0 Introduction

By letter dated January 20, 1983 the Florida Power & Light Company (FPL) submitted a request to reload and operate Unit 1 of the St. Lucie nuclear plant for Cycle 6 (Ref. 1). In support of the request, the licensee submitted a reload safety analysis report (Ref. 2), a plant transient analysis report (Ref. 3), a loss of coolant accident (LOCA) analysis report (Ref. 4), an XNB DNB correlation for St. Lucie 1 (Ref. 5), and statistical thermal margin methodology reports (Refs. 6, 7, 8, 9, and 10).

The NRC staff has reviewed the application and the supporting documents and has prepared the following evaluation of the fuel design, nuclear design, and thermal-hydraulic design of the core as well as an evaluation of those plant transients that were reanalyzed for Cycle 6. In addition, a summary and evaluation of the Technical Specification (TS) changes reviewed are also presented.

The St. Lucie Unit 1 Cycle 6 reload will consist of a combination of Exxon Nuclear Company (ENC) supplied fuel assemblies as well as previously irradiated Combustion Engineering (CE) assemblies. The Cycle 5 core consisted entirely of CE assemblies. The nominal Cycle 6 design burnup is 15,492 MWD/MTU.

2.0 FUEL DESIGN

2.1 Fuel Thermal-Mechanical Analysis

The Cycle-6 core consists of 84 fresh ENC XN-1 fuel assemblies in Batch H and a total of 133 CE fuel assemblies: 64 in Batch G, 68 in Batch F, and 1 in Batch E. The XN-1 fuel assemblies, whose design is described in the generic topical report XN-NF-82-09 (Generic Mechanical Design Report Exxon Nuclear 14X14 Fuel Assemblies for Combustion Engineering Reactors), are the first such ENC assemblies to be used in St. Lucie 1. ENC stated that there are few differences other than minor details between the XN-1 fuel assemblies and other ENC 14X14 fuel assemblies such as now residing in the Maine Yankee and Fort Calhoun reactors.

The staff has completed the review of XN-NF-82-09 and approved it for licensing applications (Ref. 11). However, there were six conditions attached to the approval of the use of the ENC 14X14 fuel design for C-E reactors. These conditions were:

1. The licensee must confirm that the design power histories shown in Tables 6.3 and 6.4 of XN-NF-82-09 bound the power limits for the application in question.

2. Following the approval of RODEX2 code, the licensee must confirm or redo the following analyses, which were reviewed on the basis of RODEX2 results: design strain, strain fatigue, external corrosion, rod pressure, and pellet cladding interaction.

3. A plant-specific analysis of rod bowing must be performed to determine an appropriate DNBR penalty.

4. Prior to the second cycle of operation, the licensee must provide an analysis using approved methods that shows no cladding creep collapse for the design lifetime of the fuel.

5. The licensee must make sure that the fuel performance code that is used to initialize Chapter 15 accident analyses has current NRC approval.

6. The licensee must address the requirements of Appendix A to SRP Section 4.2 including NUREG-0609 to show that proposed cores containing the ENC 14X14 fuel will satisfy the structural acceptance criteria.

The staff has evaluated these six conditions during the course of its review, and the staff's conclusions are described in the following paragraphs.

2.1.1 Power History

The licensee showed in the approved generic report XN-NF-82-09 that the Cycle-6 expected power history is bounded by the design power profiles described in Tables 6.3 and 6.4 of that report. The staff finds this acceptable.

2.1.2 RODEX2 -- Design Strain, Strain Fatigue, External Corrosion, Rod Pressure, and PCI Analyses

The analyses of design strain, strain fatigue, external corrosion, rod pressure, and pellet cladding interaction (PCI) were described in the approved generic report XN-NF-82-09. ENC used the new RODEX2 code for these analyses to demonstrate that the design limits on these physical parameters would not be exceeded throughout the entire lifetime. We have completed the RODEX2 review and approved it for licensing applications (Ref. 12). Since these analyses bound the Cycle-6 application, the staff concludes that these analyses are acceptable for Cycle 6.

2.1.3 Rod Bowing

ENC used their approved rod bowing methodology (Ref. 13) to analyze the magnitude of rod bow for Cycle 6. ENC calculations indicated that 50 percent closure of rod-to-rod gap occurs at an assembly exposure of about 85,000 MWd/MTU. Significant impact to MDNBR due to rod bow does not occur until gap closures beyond 50 percent. Since the Cycle-6 projected burnup is much less than 85,000 MWd/MTU and since an approved rod bow methodology is used, the staff concludes that rod bow does not limit MDNBR and the anslysis is acceptable for ENC fuel during cycle-6 operation.

The rod bow penalty for CE fuel was calculated for high burnup (44.5 GWD/MTM peak assembly) using the NRC approved interim method (Ref. 14). The resulting incremental penalty to be applied to the XNB correlation was 1.35 percent greater for CE fuel than Exxon fuel. Had the Exxon methodology been used, neither fuel type would experience a DNB penalty due to rod bow.

In the St. Lucie Cycle-6 SAR (XN-NF-82-81), the Exxon fuel assembly was found to have a 6.0 percent lower MDNBR than the limiting CE bundle for Cycle 5 due to flow diversion from the Exxon bundles to CE bundles. The Cycle-6 MDNBR for CE fuel is 2 percent higher than for Cycle 5. Therefore, the application of 1.35 percent rod bow penalty to the CE fuel rod will not make it more limiting than ENC fuel analyzed for Cycle 6. The staff concludes that the MDNBR analysis for Cycle 6 bounds operation of Exxon and CE fuels.

2.1.4 Cladding Creep Collapse

ENC developed a new approach for calculating the creep collapse for the 14X14 fuel design. The new approach is described in detail in an ENC generic topical report on high burnup fuel (XN-NF-82-06, Qualification of Exxon Nuclear Fuel for Extended Burnup), which is under review. Since cladding collapse is a phenomenon that is not a concern until late in life, it is not expected to impact the operation of ENC 14X14 fuel during Cycle 6. The projected peak rod burnup of Cycle 6 is approximately 24,000 MWd/MTU, which is significantly less than the burnup ($\sim$40,000 MWd/MTU) required to cause cladding collapse according to the previously approved COLAPX code (XN-72-23). Accordingly, the staff concludes that there is reasonable assurance that cladding collapse will not occur in ENC 14X14 fuel rods during Cycle 6 operation.

The design burnup limit of creep collapse for CE fuel assemblies is 38,000 MWd/MTU peak assembly. Since the CE fuel in Cycle 6 will go beyond that limit, the staff requested an additional analysis to demonstrate that CE fuel will not have a collapse problem during Cycle-6 operation. The licensee provided an analysis (Ref. 12) using a new ENC creep collapse method to show that the CE fuel will not experience collapse for peak assembly burnups up to 45,000 MWd/MTU. However, the new ENC method is currently under review and has not yet been approved. The staff is thus unable to conclude that the CE fuel will not have a collapse problem in Cycle 6. Accordingly, the licensee is required to use an approved method to show that the CE fuel will not experience collapse prior to reaching 38,000 MWd/MTU peak assembly unless the new ENC creep collapse methodology has been approved.

2.1.5 LOCA Initial Conditions and Cladding Swelling and Rupture

ENC used the approved steady-state code, GAPEXX (XN-73-25), rather than the new approved RODEX2 code (XN-NF-81-58) to calculate Cycle-6 LOCA initial conditions including stored energy and rod pressure for the ENC EXEM/PWR evaluation model. Since the version of the GAPEXX code that was used is also approved and includes a correction for the effects of high burnup on fission gas release, the staff finds this analysis acceptable.

The cladding swelling and rupture model in XN-NF-82-07 (Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model) was recently approved for use in the ENC ECCS evaluation model (XN-NF-82-20). Since ENC used this approved swelling and rupture model for cladding in the ECCS analysis, this portion of the analysis is acceptable.

2.1.6 Seismic-and-LOCA Loading

An analysis of the structural adequacy of the fuel assemblies in St. Lucie 1 in response to seismic-and-LOCA loadings was an initial plant requirement (see FSAR Sections 3.6 and 3.7). Such an analysis was provided for CE fuel in the FSAR.

In 1975 an additional loading due to asymmetric blowdown forces on PWRs during LOCA was identified. As a result, NRC issued NUREG-0609 (Asymmetric Blowdown Loads on PWR Primary Systems) to address this concern and required all PWRs to submit such an analysis for evaluating fuel assembly structural adequacy.

The licensee submitted analyses (Ref. 15) to address the capability of the control element assemblies, reactor internals, and fuel assemblies under asymmetric LOCA load. The analyses of the fuel indicated that for an unrestrained inlet break localized crushing of the grid spacers occurs near the top of some of the peripheral assemblies. The licensee then further analyzed these failed assemblies and demonstrated that fuel coolability of the core was maintained during the accident. Therefore, the licensee concluded that the existing design has significant capability to accommodate these postulated events.

Since the review of asymmetric blowdown analyses has not been completed for any CE reactor, including St. Lucie 1, and since the licensee has demonstrated the structural adequacy of the fuel in the FSAR under previous acceptance criteria that is still currently acceptable, the staff concludes that no additional analysis of seismic and LOCA response is required for Cycle 6.

2.2 Flux Peaking Augmentation Factors

In proposing to remove augmentation factors from Technical Specifications, ENC presented evidence using gamma scanning that no axial gaps occurred in about 500 irradiated fuel rods from Big Rock Point and Oyster Creek as described in the ENC generic high burnup report (XN-NF-82-06). However, these data were taken from BWR fuel rods, which have different geometry and pressures from PWR fuel rods, so it is not certain that the BWR conclusion applies to PWRs. Subsequently, ENC submitted profilometry measurements of 40 PWR rods, which indirectly substantiated that there were no fuel pellet axial gaps (Ref. 15). While the no gap observation is promising, it has not been made conclusively at this time and will be pursued to conclusion in our review of the Exxon high burnup report XN-NF-82-06.

The staff notes that other fuel vendors such as Westinghouse have performed detailed local power analyses of flux peaking that have been reviewed and accepted. These analyses show that flux peaking (power spiking) due to axial fuel pellet gaps does not adversely affect DNB or LOCA behavior because of the compensating effects of flux reduction in the rod with the gap. Although CE does not have such an analysis available, the similarity of the neutronic analysis to that of Westinghouse is sufficient to lead the staff to believe that the same conclusion is also applicable to a CE plant such as St. Lucie Unit 1.

Therefore, the staff concludes that it is acceptable to remove augmentation factors from Technical Specifications for Cycle 6. The staff will generally resolve the no gap observation in its review of ENC high burnup report (XN-NF-82-06) and address the resolution in our safety evaluation of the following cycle (Cycle 7).

2.3 Post-irradiation Surveillance

Since the ENC XN-1 fuel is nearly identical to earlier ENC fuel for CE reactors, the staff does not view this fuel as a new design requiring additional surveillance. However, the staff recommends that the licensee adopt a routine post-irradiation surveillance program as suggested by the SRP (NUREG-0800, p 4.2-12). Such a qualitative visual examination of discharged fuel would give indications of cladding collapse, inadequate shoulder-gap clearance, or other anomalies that are not expected.

3.0 NUCLEAR CORE DESIGN

3.1 Core Physics Characteristics

The St. Lucie Unit 1 Cycle 6 core consists of 217 fuel assemblies, each having a 14X14 fuel rod array. Of these, 84 are fresh (unirradiated) ENC assemblies with an enrichment of 3.67 w/o U-235 and 133 are previously irradiated CE assemblies. The core also contains 656 fresh Al_2O_3 - B_4C burnable absorber rods distributed among 56 of the 84 ENC supplied fuel assemblies. These rods contain 23.8 mg/in of B-10. The Cycle 6 loading pattern has been designed so as to achieve a low radial leakage which will reduce the neutron flux to critical reactor vessel longitudinal welds and thus increase the time available prior to exceeding the proposed NRC pressurized thermal shock RT-NDT screening criteria. This is achieved by scatter-loading the fresh fuel throughout the core with the fresh assemblies containing the burnable absorber rods loaded in the core interior. The exposed fuel is also scatter-loaded in the center to control power peaking. The Cycle 6 loading pattern ensures that the peak linear heat rate (LHR) will not exceed 15 kw/ft and that the integrated radial peaking factor (F_r) and the planar radial peaking factor (F_{xy}) will not exceed 1.70 in any fuel rod through the cycle under nominal full power operating conditions.

The nuclear design and safety analysis for Cycle 6 is based on a Cycle 5 burnup of 13,215 $\pm$ 500 MWD/MTU. The Cycle 6 length is predicted to be 15,492 $\pm$ 300 MWD/MTU at a core power of 2700 MWt.

3.2 Power Distributions

Hot full power (HFP) fuel assembly relative power distributions calculated for beginning-of-cycle (BOC) and end-of-cycle (EOC) conditions show that the maximum unrodded peaking factors for Cycle 6 are 1.53 for F_r and 1.56 for F_{xy} and occur at EOC. The proposed Technical Specification limit on F_r and F_{xy} is 1.70 including uncertainties and an allowance for azimuthal tilt. The BOC HFP equilibrium xenon LHR, including uncertainties of 7 percent for measurement, 3 percent for engineering, 1 percent for densification and 2 percent for thermal power, is calculated to be 12.6 kw/ft. The flux peaking augmentation factor due to axial gaps in the fuel rod pellet stack has been eliminated for the ENC fuel for the reasons discussed in Section 2. In addition, ENC neutronics calculations have shown that the product of the peak rod power for CE fuel remaining in the core and the maximum augmentation factor is less than the ENC fuel peaking factor at any location during Cycle 6. Therefore, the staff agrees that flux peaking augmentation factors need not be applied to CE fuel either. The Technical Specification limit on LHR is 15 kw/ft. Comparisons of the radial peaks and peak LHR given in the above power distributions with the allowable values shown in the Technical Specifications demonstrate the adequacy of the results given in the safety analyses.

A preliminary review of the ENC analysis of the power distribution measurement uncertainty for St. Lucie Unit 1 (Ref. 17) indicates that the one-sided 95/95 tolerance limits on the power peaking factors are within the values specified in the Technical Specifications. Specifically, these are 7.0 percent for the total peaking factor (F_Q) and 6.0 percent for F_r. The methods used appear to be appropriate and consistent with those used by other PWR fuel vendors and approved by the staff. The staff, therefore, expects that any questions which may arise during our final review of this report will not affect our approval of the continued use of the above quoted Technical Specification power distribution measurement uncertainty values.

3.3 Control Requirements

The value of the required shutdown margin is determined by the steam line break analysis and has been decreased to 3600 pcm for Cycle 6 where 1000 pcm is equivalent to 1 percent $\Delta k/k$ in reactivity. Based on this value of required shutdown margin and on calculated available scram reactivity including a maximum worth stuck control rod and appropriate calculational uncertainties, sufficient excess exists between available and required scram reactivity for all Cycle 6 operating conditions. These results are derived by approved methods and incorporate appropriate assumptions and are, therefore, acceptable.

3.4 Reactivity Coefficients

The calculated Doppler coefficients for Cycle 6 are similar to those for Cycle 5. The Technical Specifications require that the moderator temperature coefficient (MTC) be less positive than +5 pcm/°F below 70 percent of rated power, less positive than +2 pcm/°F above 70 percent power and more positive than -22 pcm/°F at rated power. Calculations have shown that these limits are met. Since approved methods have been used and appropriate values incorporated in the safety analyses, the reactivity coefficients for Cycle 6 are acceptable.

3.5 Analytical Methodology

The nuclear design methodology used in Cycle 6 is described in References 18, 19, and 20 and has been approved by the staff. The XTGPWR reactor simulator code was used to calculate physics parameters such as power distributions, control rod worths, cycle lifetime, LHR, F_r, and F_{xy}. This code has been approved by the staff for use by ENC for PWRs for which ENC provides reload fuel (Ref. 21).

4.0 THERMAL-HYDRAULIC ANALYSES

This evaluation includes a review of the thermal-hydraulic design analysis for St. Lucie Unit 1, Cycle 6. This review is necessitated by the fact that Cycle 6 will contain a mixed loading of ENC and CE fuel assemblies.

The objective of this review is to confirm that the thermal-hydraulic design of the reload has been accomplished using acceptable methods, and provides an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational transients. Besides a normal review of the Technical Specifications and reload safety analysis reports an expanded review was performed in the following areas:

(1) mixed core thermal-hydraulic design methodology;

(2) the hydraulic compatibility of ENC fuel with the existing CE fuel and the acceptability of any changes in hydraulic performance between Cycle 6 and the reference cycle;

(3) review of ENC's statistical setpoint methodology for CE plants;

(4) review of XNB, ENC's DNB Correlation for 14X14 CE type PWR fuel designs; and

(5) minimum DNBR.

4.1 Mixed Core Thermal-Hydraulic Design Methodology

The thermal-hydraulic design methodology used by ENC is comprised of two steps. Initially, a core-wide calculation is performed on an assembly-by-assembly basis. In this analysis the limiting bundle is placed at its allowable-maximum radial peak while the remaining bundles are at their nominal powers. Inlet flow maldistributions are accounted for by a reduction of 5 percent in the hot bundle flow.

The results of this calculation are the axial flow distribution for the hot assembly and the crossflow boundary conditions which will be used in the detailed subchannel model. These boundary conditions were originally stored as the average of all the boundary conditions on the hot assembly. However, during the course of our review, ENC modified their code to properly store the corewide crossflow boundary conditions. That is, they do not average the crossflow conditions but use the actual crossflows as seen by the limiting assembly.

Next, an octant of the hot assembly is modeled on a rod-by-rod basis to determine the minimum DNBR for the core. In this model, crossflow between the limiting and adjacent fuel assemblies is accounted for by using the boundary conditions stored during the corewide calculations while flow redistribution within the limiting assembly is accounted for via crossflow between adjacent subchannels. As part of their subchannel analysis, ENC increases the peak rod heat flux by typically 3 percent to account for extremes in fuel rod manufacturing tolerances and uses a flat peaking distribution within the rod array except for the limiting rod which is placed at its maximum peak.

The analytical tools which comprise the design methodology are the XCOBRA-IIIC computer code (XN-NF-75-21(P), Revision 2) and the XNB critical heat flux (CHF) correlation (XN-NF-621, Revision 1).

The methodology detailed in XN-NF-82-21(P), is the subject of a separate staff review which is described in the letter from C. Thomas (NRC) to C. Chandler (Exxon) dated August 26, 1983, "Review of XN-NF-82-21(P); Revision 1." The staff position transmitted in this memorandum is that the thermal-hydraulic design methodology presented in XN-NF-82-21(P), Revision 1 is acceptable for performing steady-state core thermal-hydraulic calculations when the proper method of storing crossflow boundary conditions is used. In addition, an adjustment of 2 percent on the minimum DNBR must be included for mixed cores containing hydraulically different fuel assemblies.

4.2 Thermal-Hydraulic Compatibility and Cycle to Cycle Comparisons

Hydraulic performance differences between CE and ENC fuel were tested with pressure drop tests performed in ENC's Hydraulic Loop Test Facility. Using the loss coefficients from these tests ENC determined that the impact of an all-ENC core on primary flow is less than 0.5 percent. Thus insertion of Exxon fuel into the St. Lucie Unit 1 reactor will not significantly impact primary coolant flow.

The licensee also was asked to compare the major Thermal Hydraulic Parameters of a reference cycle with all CE fuel and Cycle-6 to justify the differences in the principal parameters. These parameters are given in Table 4.1 with explanations given in the notes to the table.

The staff finds in accordance with our approved procedure, with an adjustment of 2 percent on the minimum DNBR to offset uncertainty in the mixed core methodology, the hydraulic differences between the ENC assemblies and CE assemblies and their effect on the major hydraulic performance parameters for Cycle-6 are acceptable.

TABLE 4.1

ST. LUCIE UNIT 1 THERMAL-HYDRAULIC PARAMETERS AT FULL POWER

General Characteristics	UNIT	Reference Cycle 4	XN-1 Cycle 6
Total Heat Output (core only)	MWt 10^6 Btu/hr	2700 9215	2700 9214.3
Fraction of Heat Generated in Fuel Rod		.975	.975
Primary System Pressure			
Nominal	psia	2250	2250
Minimum in steady state	psia	2200	2200
Maximum in steady state	psia	2300	2300
Inlet Temperature	^oF	549.	549.
Total Reactor Coolant Flow (steady state)	gpm 10^6 lb/hr	370,000 139.3	370,000 139.4
Coolant Flow through Core	10^6 lb/hr	134.1	134.8
Hydraulic Diameter (nominal channel)	in	.528 (C.E.)	.541 (ENC)(1)
Average Mass Velocity	10^6 lb/hr-ft^2	2.51	2.517
Total Pressure Drop across Vessel (based on nominal dimensions and minimum steady state flow)	psi	33.6	32.6(2)
Core Average Heat Flux (accounts for above fraction of heat generated in fuel rod and axial densification factor)	Btu/hr-ft^2	183843	187089
Total Heat Transfer Area	ft^2	48872	48500
Film Coefficient at Average Conditions	Btu/hr-ft^2-^oF	5820	5300
Average Film Temperature Difference	^oF	33	35
Average Linear Heat Rate of Undensified Fuel Rod (accounts for above fraction of heat generated in fuel rod)	kW/ft	6.14	6.25
Average Core Enthalpy Rise	Btu/lb	68.7	68.36

ST. LUCIE UNIT 1 THERMAL-HYDRAULIC PARAMETERS AT FULL POWER (Cont.)

General Characteristics	UNIT	Reference Cycle 4	XN-1 Cycle 6
Calculational Factors			
Engineering Heat Flux Factor		1.03	1.03
Total Planar Radial Peaking Factors			
For DNB Margin Analyses (F_r)		1.70	1.70
Limiting Transient Minimum DNBR		1.23 (Loss of Coolant Flow)	1.33[4] (Loss of Coolant Flow)
Minimum Allowable DNBR		1.23 (CE-1)	1.17 (XNB)

NOTES

1. The hydraulic diameter cited for Cycle 6 represents ENC fuel and reflects the ENC fuel's increased rod diameter and decreased flow area.
2. The Cycle 6 pressure drop is based on measured core data.
3. The ENC limit is a specified maximum allowable value.
4. The difference in values results from modeling and correlation differences.

4.3 Review of Exxon's Statistical Setpoint Methodology for CE Plants

The review of XN-NF-507 - "ENC Setpoint Methodology for CE Reactors, Statistical Setpoint Methodology," September 1982, is in progress and is nearing completion. Based on the review of the methodology to date, the staff has determined that it is acceptable for use in licensing the St. Lucie Unit 1 Cycle 6 reload.

EG&G Idaho, Inc., has reviewed for the NRC the ENC statistical setpoint methodology for CE reactors. This methodology has been proposed by ENC for establishing the local power density and departure from DNB protective system trip setpoints and limiting conditions for operation (LCOs) for reload cores for CE reactors. The methodology uses statistical methods and probability, or confidence limits to combine parameter and measurement uncertainties, and establish the reactor power and thermal margin/low pressure (TM/LP) setpoints, and the LCOs for allowable power. The methodology is that which is outlined and briefly described in References 6, 7, and 8, and supplemented by Reference 10, and by the responses to review questions in Reference 26.

The review of the methodology is based on review of References 6 through 10 concerning the ENC setpoint methodology, References 5 and 23, concerning the XNB DNB correlation, and on the review questions and ENC responses to the review questions in References 24 through 26. Many other references, not listed here, on engineering statistics and probability analysis, and on use of statistical methods for pressurized water reactor setpoint and limit analyses, were used in the review.

The major conclusions and findings of the EG&G review are as follow:

(1) the various methodologies used by ENC to combine input parameter and measurement uncertainties and to define a 95 percent probability with a 95 percent confidence (95/95) bound for a setpoint or LCO are technically sound and conservatively applied.

(2) the individual parameter uncertainties (both magnitude and distribution) are probably plant specific. Each parameter uncertainty and uncertainty distribution used in the statistical setpoint methodology needs to be justified and reviewed for each specific plant application; and

(3) The XNB correlation and its uncertainty have been separately reviewed for the validity and adequacy of the correlation based on its test data. But, when the correlation is applied in a statistical setpoint analysis for a specific plant, the DNB uncertainty needs to include an adequate allowance for the uncertainty in the DNB mean and its variance when applied to fuel bundle designs different from test bundles used for obtaining the DNB test data. The XNB correlation variance does include a bundle-to-bundle variance component based on test data for many different test bundle designs. Additionally the ENC application of the XNB correlation includes a conservative rod bowing penalty. Therefore, between the two of these, potential uncertainty in the DNB correlation when applied to the ENC CE plant reload fuel bundles is adequately covered. When a conservative rod bowing penalty is not used, then the DNB uncertainty will need to be reviewed for adequacy for the specific application of the DNB correlation.

The staff has reviewed the individual parameter uncertainties used for St. Lucie as recommended by EG&G. The staff has found that the parameter uncertainties submitted by FPL and used in the statistical setpoint methodology for St. Lucie Unit 1 Cycle 6 are technically sound and properly applied.

4.4 Review of XNB, ENC's DNB Correlation for CE 14X14 PWR Fuel Designs

In the memorandum from L. S. Rubenstein to F. Miraglia, "Review of XN-NF-621, Revision 1" dated March 31, 1983 it is stated that the XNB critical heat flux correlation and the 1.17 95/95 limit associated with the correlation are acceptable. However, it is also stated that if XNB is used on fuel designs not contained in the data base ENC must provide additional test data for these fuel bundles or a quantified justification of XNB's applicability to this bundle type.

St. Lucie Unit 1 contains fuel of the CE 14X14 type design which was not explicitly included in the original XNB data base. In order to confirm the XNB critical heat flux correlation for bundles of a CE 14X14 design, two sets of data measured experimentally at Columbia University for CE are analyzed by ENC.

Both sets of data include one large unheated guide tube, 21 heated rods, non-uniform axial profiles, 9 grid spacers, and a 12.5 ft heated length. Columbia refers to these sections as Section CE58 and Section CE60. These test sections were selected because they were prototypic of bundles in a reactor and included both upskew and downskew power profiles. Test section CE58 used an upskew axial profile while CE60 used a downskew profile.

Based upon a XCOBRA-3C analysis, a MDNBR was obtained for each data point within each test section. The corresponding local conditions are obtained from the subchannel output. The differences in fluid conditions for the limiting subchannel between the rod indicating CHF and the XCOBRA prediction of the rod indicating CHF were observed to be small.

The analysis of the test MDNBRs show for test section CE58 a mean MDNBR of 1.0003 with a standard deviation of 0.07595 for 71 data points and for test section CE60 a mean MDNBR of 0.9749 with a standard deviation of 0.09604 for 77 data points. ENC's statistical analysis of the data show section CE58 to have a 95/95 limit of 1.15 and CE60 to have a 95/95 limit of 1.164. Thus both test sections are individually protected by the XNB 95/95 limit of 1.17.

In its review the staff has noted (see Table 4.2) that there is a difference in the range of test conditions between the CE58, CE60 test sections and the original test conditions for the XNB correlation. The staff does not feel it is appropriate to expand the range of accepted conditions for the XNB correlation for fuel types other than the CE 14X14 design.

Thus, the staff concludes that the use of the XNB with a 95/95 limit of 1.17 for CE 14X14 fuel designs is acceptable but that the range of approved conditions for the XNB correlation shall not be expanded for fuel types other than CE 14X14 fuel design.

4.5 Minimum DNBR

The safety analyses for St. Lucie Unit 1 was performed to analytically demonstrate the maintenance of a minimum DNBR of >1.22 throughout each analyzed transient. The DNBR safety limit of 1.22 is acceptable to the staff as it conservatively bounds the 1.17 MDNBR of the XNB correlation after rod bow penalties and the 2 percent adjustment for uncertainties in mixed core methodology are applied.

4.6 Thermal-Hydraulic Evaluation Summary

The staff has reviewed the St. Lucie Unit 1 Cycle 6 reload thermal-hydraulic design and find the following:

1. Assuming the adjustment of 2 percent on minimum DNBR imposed by the mixed core methodology, the hydraulic differences between ENC assemblies and CE assemblies and their effect on the major hydraulic performance parameters for Cycle 6 are acceptable.

2. The ENC statistical setpoint methodology together with the St. Lucie Unit 1 plant-specific uncertainty parameters are acceptable for determining safety limits for St. Lucie Unit 1.

3. The XNB correlation with a 1.17 95/95 MDNBR limit may be used for CE 14X14 fuel designs but the test conditions of the 14X14 data base may not be used to extend the applicability of XNB for other than 14X14 fuel types.

4. The 1.22 MDNBR limit used by ENC in setting the plant safety limits conservatively bounds the 1.17 XNB limit and analyses uncertainties.

The staff concludes that the thermal-hydraulic design of the St. Lucie Unit 1 Cycle-6 core has been accomplished using acceptable methods and that the proposed operation provides an acceptable margin of safety from conditions which could lead to fuel damage during normal operation and anticipated operational occurrences.

TABLE 4.2

TEST CONDITIONS

Parameter	CE58, CE60	XNB Correlation
Pressure (psia)	1495-2475	1395-2425
Inlet Mass Velocity ($Mlb/hrft^2$)	0.96-4.04	0.92-3.04
Local Enthalpy (BTU/lb)	617-793	594.85-821.24
Local Quality	-.13 to +.31	-.2 to +.3
Inlet Subcooling (BTU/lb)	16-338	37.2-336.34
Heated Length (in)	150	66-168
Spacer Span (in)	17.4	14.3-26.2
Vendor	CE	ENC, CE, W
Grid Design	Non-vaned, Vaned	Non-Vaned, Vaned
Axial Profile	Upskew, Downskew	Chopped Cosine, Uniform, Upskew
Hydraulic Diameter (Nominal Channel) (in)	0.333-0.534	0.463-0.528

5.0 TRANSIENT ANALYSES

Plant transient analysis is presented in References 7 and 8 to support operation of Cycle 6 with a mixed core of 84 ENC fuel assemblies and 133 CE fuel assemblies. The criteria used in the analyses are based on protecting the specified acceptable fuel design limits (SAFDLs) for anticipated operational occurrences (AOOs). Specifically, these limits are: MDNBR (based on ENC XNB) >1.22, Local Power Density $\leq$21 kW/ft, and Pressure <2750 psia. For postulated accidents, an acceptably low level of fuel damage is demonstrated such that the requirements of 10 CFR Part 100 are met.

ENC used the PTSPWR2 computer code to reanalyze the most limiting events for Cycle 6. This code is currently being reviewed by the staff. Therefore, at our request, FPL made a comparison of the limiting transients for Cycle 6 as calculated by ENC with those for the previous cycle as calculated by CE. The loss of flow and the CEA drop events were found to be the most limiting DNB transients and the loss of load event was found to be the most limiting overpressurization event. The results of the comparison showed sufficient margin to DNB and pressure limit and also showed close agreement between the Cycle 5 and Cycle 6 results. Our evaluation of the reanalyzed transients follows:

5.1 CEA Withdrawal Event

The PTSPWR2 code was used to simulate an uncontrolled CEA withdrawal for a reactivity insertion rate of 1.63×10^{-4} $\Delta\rho$/sec from full power initial conditions. The minimum DNBR falls to 1.59 and, therefore, remains well above the 95/95 acceptance limit of 1.22 using the ENC XNB critical heat flux correlation. Although PTSPWR2 is still under review, the neutronics portion of the code consists of the standard point kinetics model to determine reactivity changes during this transient. The appropriate time-dependent feedbacks are also included. The staff, therefore, concludes that the calculations contain sufficient conservatism with respect to both assumptions and models to assure that sufficient margin exists such that fuel damage will not result from a CEA withdrawal event.

5.2 CEA Drop Event

The CEA drop event was simulated by a step decrease in total reactivity at full power. The minimum DNBR falls to 1.48 and, therefore, remains above the 95/95 acceptance limit of 1.22 using the ENC XNB critical heat flux correlation. Since the power initially decreases following the CEA drop, no reactor trip occurs and protection of the SAFDLs is provided solely by the LCOs. The staff concludes that the calculations contain sufficient conservatism with respect to both assumptions and models to assure that fuel damage will not result from a CEA drop event.

5.3 Steam Line Break Analysis

The staff has reviewed the licensee's steam line break (SLB) analysis, as analyzed by ENC and which demonstrated a large margin to fuel failure (i.e., MDNBR equal to 4.5). Previous analyses, performed by the NSSS vendor, have indicated a minimum DNBR of 1.27. The analytical methodology employed by the licensee for Cycle 6 did not model the asymmetric thermal-hydraulic and asymmetric neutronic system behavior as would occur during a SLB event. The licensee has, therefore, agreed to provide confirmatory re-analysis of this event prior to next refueling. This re-analysis will conservatively model the asymmetric physics of this event. In addition, the licensee will provide a reassessment of the limiting SLB event, in accordance with the guidance outlined in Section 15.1.5 of the Standard Review Plan - SRP (NUREG-0800). This will include analyses at full and zero power operation, with and without offsite power available, and accounting for the limiting single failure and the limiting control element assembly (CEA) withdrawn during each event. This assessment is required as a result of modifying the analytical methodology, which can alter the limiting event. The methodology employed in deriving the mixing factors (asymmetry modeling), boundary conditions, neutronics, etc. will also be documented and justified in the new submittal.

Based upon previous analyses performed by the NSSS vendor, there exists adequate margin to the limits referenced by SRP Section 15.1.5. The staff concludes that the thermal-hydraulic and neutronic changes in the new fuel design will not significantly alter the plant response as evaluated by the NSSS vendor. It is for this reason that the staff approves continued operation of St. Lucie Unit 1 for Cycle 6.

5.4 Loss of Flow Event

The licensee has submitted a re-analysis of a loss of flow event in support of Cycle 6 operation. This analysis was performed by ENC, the fuel vendor for Cycle 6. ENC analyzed this event utilizing the PTSPWR2 thermal-hydraulic transient code, which is presently under staff review. The results predicted by ENC are consistent and very similar to those predicted by the NSSS vendor. Both analyses predicted the time to MDNBR to within 0.25 seconds of each other, with a MDNBR of 1.27 by the NSSS vendor and 1.35 by ENC. The staff concludes that the thermal-hydraulic and neutronic changes in fuel design will not significantly alter the plant response as evaluated by the NSSS vendor, and that the results presented in the application will not be appreciably altered by completion of the PTSPWR2 review and methodology of its implementation. The staff, therefore, finds this analysis acceptable for Cycle 6 operation.

5.5 Loss of Load Event

The licensee has submitted a re-analysis of the loss of load event in support of Cycle 6 operation. This analysis was performed by ENC, the fuel vendor for Cycle 6. ENC analyzed this event utilizing the PTSPWR2 thermal-hydraulic transient code, which is presently under staff review. The results predicted by ENC are more limiting than those predicted by the NSSS vendor. The ENC analysis predicted a pressurization of 407 psid versus a pressurization of 372 psid as analyzed by the NSSS vendor. Both results are within 110 percent of the reactor coolant system design pressure, as required by the Standard Review Plan (NUREG-0800). The staff concludes that the changes in thermal hydraulic fuel design will not significantly alter the plant response as evaluated by the NSSS vendor. The staff, therefore, finds this analysis acceptable for Cycle 6 operation.

5.6 CEA Ejection Event

The CEA ejection event was analyzed with the approved procedures developed in the ENC Generic Rod Ejection Analysis (Ref. 27). Energy deposition in the hot fuel pellet was evaluated for BOC and EOC conditions from HZP and HFP initial conditions. The analysis of this event was found to result in energy depositions well below the regulatory limit of 280 cal/gm. An analysis of the core pressure surge associated with the CEA ejection indicates a maximum pressure of 2400 psia, well below the primary coolant system pressure limit of 2750 psia. The staff concludes that the calculations contain sufficient conservatism both in the initial assumptions and in the analytical models to ensure that primary system integrity will be maintained in the event of a CEA ejection from any operating condition.

5.7 Loss of Coolant Accident (LOCA)

The loss of coolant accident was analyzed by ENC in Reference 4. ENC used the EXEM/PWR ECCS model which is still under staff review. Although the generic review is not complete, it has progressed enough that few, if any, changes are expected when the review is complete and the staff fully expects that it will find the model in compliance with Appendix K. For some plants which use ENC fuel it has been determined that maximum safety injection is the worst assumption and that it is worse than the worst single failure assumption. Our understanding from discussions with ENC is that no single failure is not the worst case for their analysis of CE plants. Thus, the staff concludes that the use of the EXEM/PWR evaluation model is acceptable for this reload with the understanding that the licensee will be required to abide by the conclusions and concerns resulting from the staff's generic review of the EXEM/PWR model. Prior to the next reload the utility should also provide written confirmation that the worst assumption regarding single failure, including no single failure, has been considered.

6.0 TECHNICAL SPECIFICATION CHANGES

1. The shutdown margin requirement with T_{avg} >200°F has been decreased to 3.6 percent $\Delta k/k$. This is consistent with the requirements of the revised steam line break analysis. The Technical Specification affected is 3.1.1.1. Since calculated scram worths show sufficient excess exists to meet this shutdown requirements and were derived by approved methods and incorporate appropriate assumptions, the decrease in shutdown margin is acceptable.

2. The moderator temperature coefficient limits were changed to less positive than 0.7×10^{-4} $\Delta k/k/°F$ at less than 70 percent of rated thermal power and less negative than -2.8×10^{-4} $\Delta k/k/°F$ at rated thermal power. The Technical Specification affected is 3.1.1.4. These changes are incorporated in the supporting physics and safety analyses for Cycle 6 using approved methods and are, therefore, acceptable.

3. The flux peaking augmentation factors due to fuel pellet densification gaps have been removed. The Technical Specifications affected are Figure 4.2-1 and Technical Specification 4.2.1.4. The staff evaluation of the removal of these augmentation factors for ENC fuel is discussed in Section 2.2 of this report. The staff also concludes that augmentation factors need not be applied to CE fuel either, as discussed in Section 3.2 of this report. The removal of the augmentation factors for Cycle-6 is, therefore, acceptable. However, prior to startup of Cycle-7, the licensee should document the basis for removing augmentation factors from the Technical Specifications.

7.0 EVALUATION FINDINGS

The staff has reviewed the fuels, physics, and thermal-hydraulics information presented in the St. Lucie Unit 1 Cycle 6 reload reports. On the basis of our review, the staff concludes that the application is acceptable with the following restriction:

Prior to reaching 38,000 MWd/MTU peak assembly, the licensee must use an approved method to show that the CE fuel will not experience creep collapse unless the new ENC methodology has been approved by the staff.

The staff has also reviewed the transient analyses, since Cycle 6 entails a change in fuel vendor and analytical methodology. The PTSPWR2 computer code was used for much of the ENC transient analysis. Since this code is presently under staff review, the acceptability of the Cycle-6 transient analysis was based primarily on comparisons between ENC results and previously reported results by CE. In the steam line break analysis, the licensee did not model the asymmetric thermal-hydraulic and neutronic system behavior. The licensee has, therefore, agreed to provide confirmatory re-analysis of the event prior to the next re-fueling. In addition, the licensee will provide a reassessment of the limiting steam line break event in accordance with the guidance outlined in Section 15.1.5 of the Standard Review Plan (NUREG-0800).

The LOCA was analyzed by ENC using the EXEM/PWR ECCS model which is still under staff review. The review has progressed to a point where the staff can conclude that the use of the EXEM/PWR evaluation model is acceptable for this reload. However, should the staff's review of the EXEM/PWR model warrant a need for reanalysis of the LOCA, the licensee will be required to submit modified analysis in conformance wih the staff's conclusions. Prior to the next reload, the licensee should also provide justification that the worst assumption regarding single failure has been considered.

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.

Conclusion

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

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