Docket File
DCS-MS-016

NOV 03 1981

Docket No. 50-335

RECEIVED NOV 12 1981 U.S. NUCLEAR REGULATORY COMMISSION

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

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. 45 to Facility Operating License No. DPR-67 for St. Lucie Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated July 23, 1981.

The amendment changes the Technical Specifications by increasing shutdown margin requirements and steam generator pressure-low trip setpoint as a result of a main steamline break reanalysis.

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

Sincerely,

Original signed by

Christian C. Nelson, Project Manager
Operating Reactors Branch #3
Division of Licensing

CP

Enclosures:
1. Amendment No. 45 to DPR-67
2. Safety Evaluation
3. Notice of Issuance

cc: See next page

8111230017 811103
PDR ADOCK 05000335
P PDR

* See previous page for concurrence and distribution.

no legal objection as to form

OFFICE	ORB#3:DL*	ORB#3:DL*	ORB#3:DL*	RSB:DSI	AD:OR:DL	OELD	
SURNAME	PMKreutzer	CNelson/pn	RAClark	BSheron	TMNovak		
DATE	10/26/81	10/26/81	10/26/81	10/28/81	10/28/81	10/ /81	

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

DISTRIBUTION:

Docket No. 50-335

DISTRIBUTION:
Docket Files
NRC PDR
L PDR
NSIC
TERA
ORB#3 Rdg
DEisenhut
PMKreutzer-3
RAClark
CNelson
OELD
I&E-4
GDeegan-4
BScharf-10
JWetmore
ACRS-10
CMiles
RDiggs
RBallard
Chairman, ASLAB
Gary File (+4)

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

Dear Dr. Uhrig:

The Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-67 for St. Lucie Unit No. 1. This amendment consists of changes to the Technical Specifications in response to your application dated July 23, 1981.

The amendment changes the Technical Specifications by increasing shutdown margin requirements and steam generator pressure-lowwtrip setpoints as a result of a main steamline break reanalysis.

As discussed on page 2 of the enclosed Safety Evaluation yourare requested to provide an analysis of fuel failure during the main steamline break event or otherwise show that analyzed DNBR's during the event remain above the 95-95 confidence limit value. Within 30 days of receipt of this letter, please submit your schedule for providing this analysis. We expect that such an analysis can be provided within 120 days.

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

Sincerely,

Christian C. Nelson, Project Manager
Operating Reactors Branch #3
Division of Licensing

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

cc: See next page

for RAC

OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	RSB:DSI	AD:OR:DL	OELD	
SURNAME	PMKreutzer	CNelson/pn	RAClark	B. SHERON	TMNovak		
DATE	10/26/81	10/26/81	10/26/81	10/ /81	10/ /81	10/ /81	

NRC FORM 318 (10-80) NRCM 0240 **OFFICIAL RECORD COPY** USGPO: 1981—335-960

Florida Power & Light Company

cc:
Robert Lowenstein, Esquire
Lowenstein, Newman, Reis & Alexrad
1025 Connecticut Avenue, N.W.
Washington, D. C. 20036

Norman A. Coll, Esquire
McCarthy, Steel, Hector & Davis
14th Floor, First National Bank Building
Miami Florida 33131

Indian River Junior College Library
3209 Virginia Avenue
Fort Pierce, Florida 33450

Administrator
Department of Environmental Regulation
Power Plant Siting Section
State of Florida
2600 Blair Stone Road
Tallahassee, Florida 32301

Mr. Weldon B. Lewis
County Administrator
St. Lucie County
2300 Virginia Avenue, Room 104
Fort Pierce, Florida 33450

U.S. Environmental Protection Agency
Region IV Office
ATTN: EIS COORDINATOR
345 Courtland Street, N.E.
Atlanta, Georgia 30308

Mr. Charles B. Brinkman
Manager - Washington Nuclear Operations
C-E Power Systems
Combustion Engineering, Inc.
4853 Cordell Avenue, Suite A-1
Bethesda, Maryland 20014

Mr. Jack Schreve
Office of the Public Counsel
Room 4, Holland Building
Tallahassee, Florida 32304

Resident Inspector/St. Lucie
Nuclear Power Station
c/o U.S.N.R.C.
P. O. Box 400
Jensen Beach, Florida 33457

cc w/enclosure(s) and incoming
dated: 7/23/81

Bureau of Intergovernmental
Relations
660 Apalachee Parkway
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. 45
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 July 23, 1981, 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 provision 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, and by amending paragraph 2.C(2) to read as follows:

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

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 45, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 3, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 45

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.

Pages

2-4
2-5
B 2-5
3/4 1-1
3/4 1-2
3/4 3-4
3/4 3-12
3/4 3-14
3/4 4-1
B 3/4 1-1

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	$\leq$ 9.61% above THERMAL POWER, with a minimum setpoint of 15% of RATED THERMAL POWER, and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.	$\leq$ 9.61% above THERMAL POWER, and a minimum setpoint of 15% of RAT THERMAL POWER and a maximum of $\leq$ 107.0% of RATED THERMAL POWER.
3. Reactor Coolant Flow - Low (1)		
Four Reactor Coolant Pumps Operating	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*	$\geq$ 95% of design reactor coolant flow with 4 pumps operating*
4. Pressurizer Pressure - High	$\leq$ 2400 psia	$\leq$ 2400 psia
5. Containment Pressure - High	$\leq$ 3.3 psig	$\leq$ 3.3 psig
6. Steam Generator Pressure - Low (2)	$\geq$ 600 psia	$\geq$ 600 psia
7. Steam Generator Water Level -Low	$\geq$ 37.0% Water Level - each steam generator	$\geq$ 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.

TABLE 2.2-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION TRIP SETPOINT LIMITS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
9. Thermal Margin/Low Pressure (1)		
Four Reactor Coolant Pumps Operating	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.	Trip setpoint adjusted to not exceed the limit lines of Figures 2.2-3 and 2.2-4.
9a. Steam Generator Pressure Difference High (1) (logic in TM/LP)	$\leq$ 135 psid	$\leq$ 135 psid
10. Loss of Turbine -- Hydraulic Fluid Pressure - Low (3)	$\geq$ 800 psig	$\geq$ 800 psig
11. Rate of Change of Power - High (4)	$\leq$ 2.49 decades per minute	$\leq$ 2.49 decades per minute

TABLE NOTATION

(1) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(2) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(3) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(4) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER.

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 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 ± 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.

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 $\geq$ 5.0% $\Delta k/k$.

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

ACTION:

With the SHUTDOWN MARGIN < 5.0% $\Delta k/k$, 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 $\geq$ 5.0% $\Delta k/k$:

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 ± 1.0% Δk/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.

TABLE 3.3-1 (Continued)

REACTOR PROTECTIVE INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Wide Range Logarithmic Neutron Flux Monitor					
a. Startup and Operating-- Rate of Change of Power - High	4	2(d)	3	1, 2 and *	2#
b. Shutdown	4	0	2	3, 4, 5	3
12. Reactor Protection System Logic	4	2	4	1, 2*	4
13. Reactor Trip Breakers	4	2	4	1, 2*	4

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the protective system trip breakers in the closed position and the CEA drive system capable of CEA withdrawal.

#The provisions of Specification 3.0.4 are not applicable.

(a) Trip may be bypassed below 1% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 1% of RATED THERMAL POWER.

(b) Trip may be manually bypassed below 685 psig; bypass shall be automatically removed at or above 685 psig.

(c) Trip may be bypassed below 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL POWER is $\geq$ 15% of RATED THERMAL POWER.

(d) Trip may be bypassed below 10^{-4}% and above 15% of RATED THERMAL POWER; bypass shall be automatically removed when THERMAL power is $\geq 10^{-4}$% or $\leq$ 15% of RATED THERMAL POWER.

(e) Trip may be bypassed during testing pursuant to Special Test Exception 3.10.3.

(f) There shall be at least two decades of overlap between the Wide Range Logarithmic Neutron Flux Monitoring Channels and the Power Range Neutron Flux Monitoring Channels.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the protective system trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
5. CONTAINMENT SUMP RECIRCULATION (RAS)					
a. Manual RAS (Trip Buttons)	2	1	2	1, 2, 3, 4	8
b. Refueling Water Tank - Low	4	2	3	1, 2, 3	9#
6. LOSS OF POWER 4.16 kv Emergency Bus Undervoltage (Under-voltage relays)	1/Bus	1/Bus	1/Bus	1, 2, 3	9#
7. AUXILIARY FEEDWATER AUTOMATIC START Steam Generator (SG) Level Instruments	4/SG	2/SG 1/	2/SG	1, 2, 3	11

1/ 2/SG for either steam generator will start one train of AFW.

TABLE 3.3-3 (Continued)

TABLE NOTATION

(a) Trip function may be bypassed in this MODE when pressurizer pressure is < 1725 psia; bypass shall be automatically removed when pressurizer pressure is ≥ 1725 psia.

(b) An SIAS signal is first necessary to enable CSAS logic.

(c) Trip function may be bypassed in this MODE below 685 psig; bypass shall be automatically removed at or above 685 psig.

The provisions of Specification 3.0.4 are not applicable.

ACTION STATEMENTS

ACTION 8 - With the number of OPERABLE channels one less than the Total Number of Channels, restore the inoperable channel to OPERABLE status within 48 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

ACTION 9 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in either the bypassed or tripped condition within 1 hour. For the purposes of testing and maintenance, the inoperable channel may be bypassed for up to 48 hours from time of initial loss of OPERABILITY; however, the inoperable channel shall then be either restored to OPERABLE status or placed in the tripped condition.

b. Within one hour, all functional units receiving an input from the inoperable channel are also placed in the same condition (either bypassed or tripped, as applicable) as that required by a. above for the inoperable channel.

c. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 48 hours while performing tests and maintenance on that channel provided the other inoperable channel is placed in the tripped condition.

TABLE 3.3-3 (Continued)

TABLE NOTATION

ACTION 10 - With the number of OPERABLE channels one less than the Total Number of Channels, operation may proceed provided the inoperable channel is placed in the bypassed condition and the Minimum Channels OPERABLE requirement is demonstrated within 1 hour; one additional channel may be bypassed for up to 2 hours for surveillance testing per Sepcification 4.3.2.1.1.

ACTION 11 - Instrument operability requirements are contained in the Reactor Protection System requirements for Reactor Trip on Steam Generator Level. If an Automatic Start channel is inoperable, operation may continue provided that the affected pump is verified to be OPERABLE per Specification 4.7.1.2.a within 8 hours and at least once per 7 days thereafter; and the Automatic Start channel shall be restored to OPERABLE status within 30 days or the reactor shall be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 3.3-4

ENGINEERED SAFETY FEATURE ACTUATION SYSTEM INSTRUMENTATION TRIP VALUES

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. SAFETY INJECTION (SIAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 5 psig	≤ 5 psig
c. Pressurizer Pressure - Low	≥ 1600 psia	≥ 1600 psia
2. CONTAINMENT SPRAY (CSAS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure -- High-High	≤ 10 psig	≤ 10 psig
3. CONTAINMENT ISOLATION (CIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Containment Pressure - High	≤ 5 psig	≤ 5 psig
c. Containment Radiation - High	≤ 10 R/hr	≤ 10 R/hr
d. SIAS	----- (See FUNCTIONAL UNIT 1 above) ----	
4. MAIN STEAM LINE ISOLATION (MSIS)		
a. Manual (Trip Buttons)	Not Applicable	Not Applicable
b. Steam Generator Pressure - Low	≥ 585 psig	≥ 585 psig
5. CONTAINMENT SUMP RECIRCULATION (RAS)		
a. Manual RAS (Trip Buttons)	Not Applicable	Not Applicable
b. Refueling Water Tank - Low	48 inches above tank bottom	48 inches above tank bottom

3/4.4 REACTOR COOLANT SYSTEM

REACTOR COOLANT LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1 Four reactor coolant pumps shall be in operation.

APPLICABILITY: As noted below, but excluding MODE 6.

ACTION:

MODES 1 and 2:

With less than four reactor coolant pumps in operation, be in at least HOT STANDBY within 6 hours.

MODES 3, 4 and 5:

Operation may proceed provided (1) at least one reactor coolant loop is in operation with an associated reactor coolant pump or shutdown cooling pump and (2) the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is maintained at $\geq$ 5.0% $\Delta k/k$ during operation in MODE 3 when less than four reactor coolant pumps are in operation. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.1 The Flow Dependent Selector Switch shall be determined to be in the 4 pump position within 15 minutes prior to making the reactor critical and at least once per 12 hours thereafter.

All reactor coolant pumps and shutdown cooling pumps may be de-energized for up to 1 hour, provided no operations are permitted which could cause dilution of the reactor coolant system boron concentration.

SAFETY VALVES - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 PSIA $\pm$ 1%.

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE shutdown cooling loop into operation.

SURVEILLANCE REQUIREMENTS

4.4.2 The pressurizer code safety valve shall be demonstrated OPERABLE per Surveillance Requirement 4.4.3.

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 5.0% $\Delta k/k$ 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 transients resulting from any postulated accident are minimal and a 1% $\Delta k/k$ shutdown margin provides adequate protection.

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 $+ 0.5 \times 10^{-4}$ $\Delta k/k/°F$ 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 $- 2.2 \times 10^{-4}$ $\Delta k/k/°F$ 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.

BASES

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 1.0% $\Delta k/k$ 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. DPR-67

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT, UNIT NO. 1

DOCKET NO. 50-335

Introduction:

Due to TMI related changes in the operation of St. Lucie Unit 1 (plant), specifically automatic initiation of Auxiliary Feedwater and manual tripping of reactor coolant pumps (RCP) on safety injection, Florida Power & Light Company (FPL or the licensee) has reanalyzed the Main Steamline Break (MSLB) event. FPL's analysis and associated Technical Specification changes were submitted on July 23, 1981. Additional information was provided by FPL's letters dated September 4 and 11 and October 20, 1981. We have evaluated FPL's submittals.

Evaluation:

I. Analysis

The MSLB event is analyzed to assure that the primary coolant system can be maintained in safe status for a range of steamline breaks. We used the criteria of Standard Review Plan section 15.1.5 in evaluating the MSLB analysis. Conservative assumptions for core burnup, scram characteristics, core flow, loss of offsite power, power level, and the worst single active component failure are required.

The MSLB event was analyzed at both hot zero power and 2754 Mwt (Ref. 1&4). The currently authorized maximum reactor core steady state power level is 2560 Mwt. In a separate action FPL, on November 14, 1980, requested authorization to operate the plant at a stretch power level of 2700 Mwt. The subject MSLB analysis, which assumes an initial power level of 2754 Mwt (102% of 2700 Mwt), updated the stretch power submittal. Although we have not yet approved the stretch power request, this MSLB analysis envelopes and is acceptable for operation at 2560 Mwt.

The MSLB analyses assume that the reactor coolant pumps are tripped upon receipt of the low pressure (1578 psi) safety injection actuation signal. This is in accordance with TMI action plan requirements and we find that this is conservative.

Fuel and core characteristics are based on conservative end-of-cycle values with the most negative moderator coefficient of reactivity allowed. This maximizes the potential for return-to-power following the primary system cooldown initiated by the steamline break. In addition, the most reactive control element assembly (CEA) was assumed to be stuck in the fully withdrawn position.

Main feedwater flow is isolated 60 seconds after the steamline break, and auxiliary feedwater flow is initiated 180 seconds after the safety injection actuation signal. Normally, main feedwater is isolated automatically following the trip signals, so the delayed isolation assumed permits more heat transfer from the primary system, which is conservative.

The results for the full power analysis show that the pressures in the primary and secondary systems do not exceed 110% of design pressure. The peak post-trip power was calculated to rise to 14% of rated power. The minimum departure from nucleate boiling ratio (DNBR) experienced during this event was 1.27 (Ref. 2). This led to 0.5% predicted fuel failure, with resultant dosage well below the 10 CFR 100 guidelines.

FPL has provided analyses of the MSLB event for St. Lucie Unit 1. The assumptions, methods, and results provided in the analyses are in conformance with SRP Section 15.1.5. We conclude that the MSLB analyses and associated operating restrictions (discussed below) are acceptable for St. Luice Unit 1.

II. Technical Specifications

A. Shutdown Margin for Modes 1, 2, 3 and 4

The shutdown margin for $T_{ave} > 200^{o}F$ has been increased from 3.3 percent $\Delta k/k$ for Cycle 4 to 5.0 percent $\Delta k/k$ for Cycle 5 to yield acceptable consequences from a steamline break event initiated at no load conditions. The staff has reviewed the CEA reactivity worths and allowances for Cycle 5 presented in Reference 3 and concludes that sufficient CEA worth is available for this required shutdown margin. This change is, therefore, acceptable. The pages affected are 3/4 1-1, B3/4 1-1 and 3/4 4-1.

B. Surveillance Requirements for Shutdown Margin in Modes 3 or 4

During the calculation of shutdown margin for modes 3 and 4, FPL proposes that the highest reactivity worth CEA need not be assumed to be stuck in the fully withdrawn position if all CEAs are verified to be fully inserted. This change will only exempt the stuck rod assumption when complying with the surveillance requirements of paragraph e of Technical Specification 3.1.1.1 (for modes 3 and 4) upon verification of all CEAs fully inserted. Once CEA withdrawal has commenced during reactor startup, the stuck CEA penalty will be included in the calculation of shutdown margin and the RCS boron concentration will be determined accordingly. We find this acceptable. The page affected is 3/4 1-2.

C. Steam Generator Pressure - Low Trip and Bypass Setpoints

To minimize the consequences of a MSLB event the setpoint for the steam generator pressure - low trip has been increased from ≥ 500 psia to > 600 psia. This would cause CEA's to drop into the core earlier in the accident. The bypass setpoint of 585 psig has been increased to 685 psig to be consistent with the new trip value. The revised trip setpoint is consistent with the MSLB analysis and is acceptable. The pages affected are 2-4 (Table 2.2-1), 2-5 (Table 2.2-1 Notation), B2-5, 3/4 3-4 (Table 3.3-1 Notation), 3/4 3-12 (Table 3.3-3 Notation) and 3/4 3-14 (Table 3/3-4).

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) because the amendment does not involve a significant increase in the probability 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.

Date: November 3, 1981

References

1. FPL letter to D. Eisenhut, NRC, from R. E. Uhrig, dated July 23, 1981.
2. FPL letter to R. A. Clark, NRC, from R. E. Uhrig, dated September 4, 1981.
3. FPL letter to R. A. Clark, NRC, from R. E. Uhrig, dated September 11, 1981.
4. FPL letter to D. G. Eisenhut, NRC, from R. E. Uhrig, dated October 20, 1981.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-335

FLORIDA POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO
FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 45 to Facility Operating License No. DPR-67, issued to Florida Power & Light Company (the licensee), which revised the Technical Specifications for operation of the St. Lucie Plant, Unit No. 1 (the facility), located in St. Lucie County, Florida. The amendment is effective as of the date of issuance.

This amendment changes the Technical Specifications by increasing shutdown margin requirements and steam generator pressure-low trip setpoints as a result of a main steamline break reanalysis.

The application for the amendment complies 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 51.5(d)(4) an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 23, 1981, (2) Amendment No. 45 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 Licensing.

Dated at Bethesda, Maryland this 3rd day of November, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing