St. Lucie Unit 1 and Unit 2
Docket Nos. 50-335 and 50-389
Proposed License Amendments
LPSI System Risk Informed AOT Extension

L-99-079, ATTACHMENT 3

ST. LUCIE UNIT 1 MARKED-UP TECHNICAL SPECIFICATIONS PAGES

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ECCS SUBSYSTEMS - $T_{avg} \geq 325°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high-pressure safety injection (HPSI) pump,

b. One OPERABLE low-pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal.

APPLICABILITY: MODES 1, 2 and 3*.

ACTION:

~~a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.~~

REPLACE

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

*With pressurizer pressure $\geq$ 1750 psia.

a.1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

a.2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS (Continued)

e. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on a Safety Injection Actuation Signal.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Signal;

   a. High-Pressure Safety Injection Pump.

   b. Low-Pressure Safety Injection Pump.

DELETE

3. Verifying that upon receipt of an actual or simulated ~~Sump~~ Recirculation Actuation Signal: each low-pressure safety injection pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.

f. By verifying that each of the following pumps develops the specified total developed head on recirculation flow when tested pursuant to the Inservice Testing Program.

1. High-Pressure Safety Injection pumps: greater than or equal to 2571 ft.

2. Low-Pressure Safety Injection pumps: greater than or equal to 350 ft.

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the RCS safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration and pressure ensure that the assumptions used for safety injection tank injection in the accident analysis are met.

The limit of 72 hours for operation with an SIT that is inoperable due to boron concentration not within limits, or due to the inability to verify liquid volume or cover-pressure, considers that the volume of the SIT is still available for injection in the event of a LOCA. If one SIT is inoperable for other reasons, the SIT may be unable to perform its safety function and, based on probability risk assessment, operation in this condition is limited to 24 hours.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double ended break of the largest RCS cold leg pipe downward. In addition, each ECCS subsystem provides long term core cooling capability in the recirculation mode during the accident recovery period.

INSERT NEW PARAGRAPH

The Surveillance Requirements provided to ensure OPERABILITY of each component ensure that at a minimum, the assumptions used in the accident analyses are met and that subsystem OPERABILITY is maintained.

The limitations on HPSI pump operability when the RCS temperature is ≤ 270°F and ≤ 236°F, and the associated Surveillance Requirements provide additional administrative assurance that the pressure/temperature limits (Figures 3.4-2a and 3.4-2b) will not be exceeded during a mass addition transient mitigated by a single PORV. A limit on the maximum number of operable HPSI pumps is not necessary when the pressurizer manway cover or the reactor vessel head is removed.

TS 3.5.2, ACTION a.1. provides an allowed outage/action completion time (AOT) of up to 7 days from initial discovery of failure to meet the LCO provided the affected ECCS subsystem is inoperable only because its associated LPSI train is inoperable. This 7 day AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk-informed" AOT extension. Entry into this ACTION requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP) which is described in the Administrative Procedure (ADM-17.08) that implements the Maintenance Rule pursuant to 10 CFR 50.65.

St. Lucie Unit 1 and Unit 2
Docket Nos. 50-335 and 50-389
Proposed License Amendments
LPSI System Risk Informed AOT Extension

L-99-079, ATTACHMENT 4

ST. LUCIE UNIT 2 MARKED-UP TECHNICAL SPECIFICATIONS PAGES

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EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS SUBSYSTEMS - T_{avg} GREATER THAN OR EQUAL TO 325°F

LIMITING CONDITION FOR OPERATION

3.5.2 Two independent Emergency Core Cooling System (ECCS) subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE high pressure safety injection pump,

b. One OPERABLE low pressure safety injection pump, and

c. An independent OPERABLE flow path capable of taking suction from the refueling water tank on a Safety Injection Actuation Signal and automatically transferring suction to the containment sump on a Recirculation Actuation Signal, and

d. One OPERABLE charging pump.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

~~a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.~~

REPLACE

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

*With pressurizer pressure greater than or equal to 1750 psia.

a.1. With one ECCS subsystem inoperable only because its associated LPSI train is inoperable, restore the inoperable subsystem to OPERABLE status within 7 days or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

a.2. With one ECCS subsystem inoperable for reasons other than condition a.1., restore the inoperable subsystem to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS (Continued)

2. A visual inspection of the containment sump and verifying that the subsystem suction inlets are not restricted by debris and that the sump components (trash racks, screens, etc.) show no evidence of structural distress or corrosion.

3. Verifying that a minimum total of 173 cubic feet of solid granular trisodium phosphate dodecahydrate (TSP) is contained within the TSP storage baskets.

4. Verifying that when a representative sample of 70.5 ± 0.5 grams of TSP from a TSP storage basket is submerged, without agitation, in 10.0 ± 0.1 gallons of 120 ± 10°F borated water from the RWT, the pH of the mixed solution is raised to greater than or equal to 7 within 4 hours.

f. At least once per 18 months, during shutdown, by:

1. Verifying that each automatic valve in the flow path actuates to its correct position on SIAS and/or RAS test signals.

2. Verifying that each of the following pumps start automatically upon receipt of a Safety Injection Actuation Test Signal:

a. High-Pressure Safety Injection pump.

b. Low-Pressure Safety Injection pump.

DELETE

3. Verifying that upon receipt of an actual or simulated ~~Sump~~ Recirculation Actuation Signal: each low-pressure safety injection pump stops, each containment sump isolation valve opens, each refueling water tank outlet valve closes, and each safety injection system recirculation valve to the refueling water tank closes.

g. By verifying that each of the following pumps develops the specified total developed head on recirculation flow when tested pursuant to the Inservice Testing Program:

1. High-Pressure Safety Injection pumps: greater than or equal to 2854 ft.

2. Low-Pressure Safety Injection pump: greater than or equal to 374 ft.

h. By verifying the correct position of each electrical and/or mechanical position stop for the following ECCS throttle valves:

1. During valve stroking operation or following maintenance on the valve and prior to declaring the valve OPERABLE when the ECCS subsystems are required to be OPERABLE.

BASES

3/4.5.1 SAFETY INJECTION TANKS

The OPERABILITY of each of the Reactor Coolant System (RCS) safety injection tanks ensures that a sufficient volume of borated water will be immediately forced into the reactor core through each of the cold legs in the event the RCS pressure falls below the pressure of the safety injection tanks. This initial surge of water into the core provides the initial cooling mechanism during large RCS pipe ruptures.

The limits on safety injection tank volume, boron concentration, and pressure ensure that the assumptions used for safety injection tank injection in the safety analysis are met.

The safety injection tank power-operated isolation valves are considered to be "operating bypasses" in the context of IEEE Std. 279-1971, which requires that bypasses of a protective function be removed automatically whenever permissive conditions are not met. In addition, as these safety injection tank isolation valves fail to meet single failure criteria, removal of power to the valves is required.

The limit of 72 hours for operation with an SIT that is inoperable due to boron concentration not within limits, or due to the inability to verify liquid volume or cover-pressure, considers that the volume of the SIT is still available for injection in the event of a LOCA. If one SIT is inoperable for other reasons, the SIT may be unable to perform its safety function and, based on probability risk assessment, operation in this condition is limited to 24 hours.

3/4.5.2 and 3/4.5.3 ECCS SUBSYSTEMS

The OPERABILITY of two separate and independent ECCS subsystems ensures that sufficient emergency core cooling capability will be available in the event of a LOCA assuming the loss of one subsystem through any single failure consideration. Either subsystem operating in conjunction with the safety injection tanks is capable of supplying sufficient core cooling to limit the peak cladding temperatures within acceptable limits for all postulated break sizes ranging from the double-ended break of the largest RCS hot leg pipe downward. In addition, each ECCS subsystem provides long-term core cooling capability in the recirculation mode during the accident recovery period.

TS 3.5.2, ACTION a.1. provides an allowed outage/action completion time (AOT) of up to 7 days from initial discovery of failure to meet the LCO provided the affected ECCS subsystem is inoperable only because its associated LPSI train is inoperable. This 7 day AOT is based on the findings of a deterministic and probabilistic safety analysis and is referred to as a "risk-informed" AOT extension. Entry into this ACTION requires that a risk assessment be performed in accordance with the Configuration Risk Management Program (CRMP) which is described in the Administrative Procedure (ADM-17.08) that implements the Maintenance Rule pursuant to 10 CFR 50.65.

ADD NEW PARAGRAPH

50-335/389
3/13/2000

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Docket: 05000335
Docket: 05000389

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

FPL

March 13, 2000

L-2000-57
10 CFR 50.4

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Requested Corrections/Clarifications to NRC Safety Evaluation
For LPSI AOT Extension License Amendments 164 and 106

During the implementation review of the NRC Safety Evaluation (SE) for Unit 1 and Unit 2 license amendments 164 and 106, respectively, dated February 15, 2000, Florida Power & Light Company (FPL) identified several items in the SE that were not consistent with the FPL license amendment application. The SE was for the low pressure safety injection (LPSI) allowed outage time (AOT) extension. As requested by the NRC Project Manager, the inconsistencies are identified in the attachment to this letter. The NRC is requested to review the FPL comments and correct/clarify the SE as appropriate.

By letter L-2000-49 dated February 21, 2000, FPL requested an extension of the implementation period for these license amendments to 60 days from the date of receipt. The extension is needed to allow time for FPL to complete the procedure development and training associated with the implementation of the Configuration Risk Management Program (CRMP) required by this amendment. This letter confirms a telephone approval of the extension by the NRC Project Manager for St. Lucie, Kahtan Jabbour, as discussed with George Madden of my staff on February 28, 2000.

Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

Attachment

RSK/GRM

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

003693679

A001

FPL Comments on LPSI AOT Extension
Staff Safety Evaluation St. Lucie Unit 1 and Unit 2
License Amendments 164 and 106 Dated February 15, 2000

<u>SE Section 4.1:</u>

"The primary role of the low pressure safety injection (LPSI) system during power operation is to contribute to the mitigation of a large loss of coolant accident (LOCA). The postulated frequency of a large LOCA event is on the order of 10^{-4} per year. In contrast, during Modes 5 and 6, the operability of at least one LPSI train operating in the shutdown cooling mode is required at all times for reactor coolant system (RCS) heat removal. Thus, in the broad view, performing preventive and corrective maintenance at power on the LPSI system can contribute to an overall enhancement of plant safety by increasing the availability of at least one LPSI train for shutdown cooling (SDC) when it is needed in Modes 5 and 6."

<u>FPL Comment 1</u>

Staff states the large LOCA frequency is on the order of E-4/yr. The proposed license amendment (PLA) states "on the order of E-5/yr." (see below)
Staff says SDC in Modes 5 and 6, PLA stated "Modes 4, 5, and 6." (see below)

<u>*Applicable FPL PLA section 3.2*</u>

"In the upper operating modes, LPSI trains must be available in the event that LOCA mitigation becomes necessary. The estimated frequency of a large LOCA is on the order of E-05 per year. The LPSI system would also be used for RCS heat removal in the event of a SGTR or other non-LOCA design basis events, which have estimated frequencies on the order of E-03 per year and lower. In contrast, at least one LPSI train is required to be operable for RCS heat removal during normal shutdown operations in Modes 4, 5, and 6, and is almost always in operation when in these modes. Therefore, in the broad view, performing preventive and corrective maintenance on LPSI trains when at power can enhance overall plant safety by increasing the availability and reliability of the LPSI system for normal shutdown-cooling operations, i.e., when it is most often needed."

<u>SE section 4.2 first paragraph</u>

"The two trains of the LPSI system, in combination with the two trains of the high pressure safety injection (HPSI) system, form two redundant ECCS trains. The two LPSI pumps are high volume, low head centrifugal pumps designed to supplement the SIT inventory in reflooding the reactor vessel to ensure core cooling during the early stages of a large break LOCA. The LPSI pumps take suction from the refueling water storage tank (RWST),

during the injection phase of a LOCA event, and pump the water through a common discharge header. Once inside containment, the LPSI headers combine with HPSI and SIT discharge piping, and flow is directed through independent injection headers into each of the four reactor coolant system (RCS) cold legs and into the reactor vessel. The LPSI system pumps start and valves open upon receipt of a safety injection actuation signal. When the **RWST** level is drawn down by inventory transfer during the injection phase, a low **RWST** level actuates a recirculation actuation signal which stops the LPSI pumps. This step is necessary to ensure adequate net positive suction head remains available for the HPSI pumps and the containment spray pumps. By design, post-LOCA long term core cooling is supplied by the HPSI pumps and containment spray pumps taking suction from the containment sump."

FPL Comment 2

Staff refers to "RWST" and the PLA refers to the "RWT." (See below)
Staff states "LPSI injects through a common header." This is not correct for Unit 2. The PLA describes the difference between Unit 1 and Unit 2. (See below).

Applicable FPL PLA section 2.1

*"Each LPSI train contains a high volume, low head, centrifugal pump designed to supplement the Safety Injection Tank (SIT) inventory in re-flooding the reactor vessel with borated water during the early stages of a large break loss of coolant accident (LOCA). The LPSI system is actuated by an automatic or manually initiated Safety Injection Actuation Signal (SIAS) which starts the associated pump and causes the LPSI flow control valves to open. The LPSI pumps transfer borated water from the **Refueling Water Tank (RWT)**, through the LPSI header(s), and into the safety injection penetrations to the Reactor Coolant System (RCS) cold legs. During the recirculation phase of the LOCA scenario, the LPSI pumps are stopped by an automatic or manually initiated Recirculation Actuation Signal (RAS) and long term core cooling is supplied by the HPSI pumps taking suction from the containment sump. **The LPSI systems for both St. Lucie units are functionally the same, but contain differences in the piping arrangement, e.g., PSL1 has a common LPSI header which branches out to each of the four high pressure cold leg penetrations whereas PSL2 has two independent LPSI headers, each branching out to two of the high pressure cold leg penetrations."***

SE section 4.2 third paragraph

"In the event that one LPSI train is out of service and the second LPSI train fails, the operator can continue to control the plant during an SGTR event by drawing steam off of the unaffected steam generator. Even though loss of both LPSI trains is beyond the design basis accident assumptions, this cooling mechanism can be maintained indefinitely, provided condensate is available to the unaffected steam generator. **Without considering condensate storage tank replenishment, St. Lucie, Units 1 and 2, have a sufficient inventory to steam the**

unaffected steam generator for more than 24 hours. St. Lucie; Units 1 and 2, also have the ability to realign the containment spray pumps to provide RCS SDC capability. Therefore, having one LPSI train out of service should not affect the licensee's ability to mitigate an SGTR event, including conditions beyond design basis."

FPL Comment 3

Both units do not have sufficient condensate inventory to steam for 24 hours without makeup. Unit 2 has sufficient volume, but Unit 1 does not. (See below)

Applicable FPL PLA section 3.2.1

"Table 6.2.1-1 of CE NPSD-995 provides a comparison of secondary side heat removal capabilities for CEOG plants, and includes the approximate condensate storage depletion time (without refill). The minimum contained volume of condensate required by the PSL1 and PSL2 TS is 116,000 gallons and 307,000 gallons, respectively. However, the steam generator heat sink can be maintained indefinitely provided make-up condensate remains available to the Condensate Storage Tank (CST). Plant procedures provide instructions for replenishing condensate inventory storage, and also include instructions for supplying the PSL1 Auxiliary Feedwater Pumps from the PSL2-CST in the event that the smaller PSL1-CST becomes unavailable. Extending the LPSI AOT would not impact this defense-in-depth capability."

FP Position Supported by SE section 4.3.3:

The licensee re-evaluated all offsite power recovery cases for both St. Lucie units. One case was added to the Unit 1 analysis for recovery of offsite power in 9 hours (approximately 1 hour before the Unit 1 condensate storage tank (CST) would deplete without condensate replenishment).

SE section 4.3.2

"The LPSI preventive and corrective maintenance (staff-estimated) weighted average single AOT risks for St. Lucie, Units 1 and 2, are 8.74E-08 for Unit 1 and 8.36E-08 for Unit 2, and are less than the acceptance guideline value 5.0E-07 from Regulatory Guide (RG) 1.177. In addition, **the change in the St. Lucie, Units 1 and 2, updated baseline core damage frequency (CDF) (as reported in the CEOG Joint Application Report) due to the LPSI AOT change is about 3%, i.e., from 2.14E-05 per year for Unit 1 and 2.35E-05 per year for Unit 2 to 2.2E-05 per year for Unit 1 and 2.4E-05 per year for Unit 2.** The change in CDF of 6E-07 per year for Unit 1 and 5E-07 per year for Unit 2 is within the acceptance guidelines published in RG 1.174. The staff-estimated weighted average incremental conditional large early release probabilities (ICLERPs) are 2.12E-09 for Unit 1 and 1.46E-09 for Unit 2, assuming a baseline early containment failure probability (ECFP) of 0.01.

Corresponding ICLERPs for an ECFP of 0.1 are 9.95E-09 for Unit 1 and 8.95E-09 for Unit 2. All of these ICLERP values are within the RG 1.177 guideline of 5.0E-08."

FPL Comment 4

The Staff references risk assessment results provided in the original CEOG Joint Applications report but states/implies these are based on the St. Lucie updated baseline CDF. As discussed in the PLA (see below), the input to the CEOG report was based on the Internal Plant Events (IPE) results and the PLA results are based on updated models. The Staff in section 4.3.3, however, repeats the PLA words regarding use of the IPE for the CEOG report and updated models for the PLA.

Applicable FPL PLA section 3.2.2

"The considerations, assumptions, methodologies, and detailed results of the initial risk assessment are reported in CE NPSD-995, Joint Applications Report for Low Pressure Safety Injection System AOT Extension, Final Report CEOG Task 836, prepared for the CE Owners Group, May 1995, as supplemented by the associated RAI response dated May 31, 1996 (CEOG Letter 96-254, D.F. Pilmer to C.I. Grimes, Chief, Technical Specifications Branch, NRR, Project No. 692; June 14, 1996). **CE NPSD-995 also contains other generic information relevant to the proposed AOT extension that is applicable to both St. Lucie units. The joint applications report, as supplemented, in conjunction with the improved data and PSA model enhancements that have been incorporated subsequent to 1995 as described in the following paragraphs, forms the risk-informed justification/basis for the proposed license amendments.**

The St. Lucie contribution to the 1995 preparation of CE NPSD-995 was generated using the IPE models developed in response to Generic Letter (GL) 88-20, *Individual Plant Examination for Severe Accident Vulnerabilities,* and associated supplements.......

Since then, FPL has updated both the models and the reliability/unavailability databases for St. Lucie Units 1 and 2. The updated models and databases were then used to re-calculate the risk numbers for the units."

SE section 4.3.2

"The Tier 3 requirements for configuration risk management are considered to be adequately satisfied, since the licensee has an on-line PRA-based monitor, called the **Safety Monitor**, to analyze the risk impact of outage configurations in a timely manner. **Procedures related to use of the Safety Monitor are St. Lucie, Units 1 and 2, Plant Administrative Procedure, ADM-17.08, "Implementation of 10 CFR 50.65, the Maintenance Rule."** The licensee has proposed adding TS Bases 3/4.5.2 and B 3/4.5.3, "ECCS SUBSYSTEMS," to provide a

means of implementing and controlling their Tier 3 process. The licensee and the staff have agreed to implementation of the Configuration Risk Management Program (CRMP as described below......."

FPL Comment 5

The PLA does not refer to our risk assessment tool as the "Safety Monitor." The PLA refers to it as the "On-line Risk Monitor." (see below)
The PLA states that we propose to include the description of the CRMP in our maintenance rule procedure. The SE implies that the procedure already has references to use of a "Safety Monitor."

Applicable FPL PLA section 3.2.3:

.... The primary tool for performing CRMP risk assessments for each St. Lucie unit will be the PSA-informed On-Line Risk Monitor (OLRM)......

Applicable FPL PLA section 3.2.4:

FPL proposes to include the description of the CRMP and its essential elements in the St. Lucie Plant Administrative Procedure (ADM) that ensures compliance with the Maintenance Rule (currently identified as ADM-17.08, Implementation of 10 CFR 50.65, the Maintenance Rule).

SE section 4.3.2 - CRMP Key Element 1

"**The intent of the CRMP is to implement Maintenance Rule, Section 50.65(a)(4) of 10 CFR** with respect to on-line maintenance for risk-informed technical specifications, with the following additions and clarifications:"

FPL Comment 6

The intent of the CRMP as committed to in the PLA is NOT to implement 10 CFR 50.65(a)(4) of the maintenance rule but to implement 10 CFR 50.65(a)(3) of the maintenance rule (see below). RG 1.177, which provides the requirements of the CRMP, also states that the intent is to implement 10 CFR 50.65(a)(3). 10 CFR 50.65(a)(4) is NOT in effect yet, therefore FPL DOES NOT know the final scope, and thus FPL has NOT committed to it at this time. The work done to date for development of the CRMP (including proposed procedure changes and OLRM) DOES NOT ensure compliance with 10 CFR 50.65(a)(4), just the CRMP.

Applicable FPL PLA section 3.2.4.2:

"Key Component 1, Implementation of CRMP: The intent of the CRMP is to implement Section a(3) of the Maintenance Rule with respect to on-line maintenance for risk-informed TS"

FPL Position Supported by RG 1.177

"2.3.7.2 Key Components of the CRMP. The licensee should ensure that the CRMP contains the following key components. Key Component 1: Implementation of CRMP
The intent of the CRMP is to implement Section a(3) of the Maintenance Rule (10 CFR 50.65) with respect to on-line maintenance for risk-informed TS, with the following additions and....."

SE section 4.3.2

FPL Comment 7

The SE only refers to what the CRMP requirements are as stated by general RG 1.177 descriptions. The PLA is more detailed on how we proposed to comply. Do we only have to address the program as stated in the SE or meet our more detailed description as stated in the PLA?

SE section 4.3.3

"Reference 5, section 5.2, and the discussion of Reference 6 in **part b., above**, provide a summary of the original IPE model peer review process."

"The licensee has updated both the models and the reliability/unavailability databases for St. Lucie Units 1 and 2. The updated models and databases were then used to re-calculate the risk numbers in support of the requested St. Lucie LPSI AOT extension. The significant model and data changes are summarized in Section 3.2.2 of the St. Lucie proposed license amendment (Reference 1) and **in part b., above**. As discussed in Reference 1, outside peer review was not performed for the update because changes that were implemented are not extensive. One or more licensee PSA engineers implemented the changes, and a licensee PSA engineer not involved with implementation of the changes performed an independent review."

"Description of PRA Quality Assurance methods."

"As noted in **paragraph b. above** and in Reference 1, the models used in the licensee's analyses were generated using the IPE models developed in response to GL 88-20, Individual Plant Examination for Severe Accident Vulnerabilities..."

FPL Comment 9

There are some references to a "part b. above." FPL could not find this "part b." It may be an incorrect reference.

SE section 4.3.5:

"The licensee **has** implemented a risk-informed Configuration Risk Management Program to assess the risk associated with the removal of equipment from service during the proposed LPSI AOT. The program **provides** the necessary assurances that appropriate assessments of plant risk configurations using the **Safety Monitor**, augmented by additional analysis, when appropriate, are sufficient to support the present AOT extension requests for the LPSI system (Tier 3)."

FPL Comment 10

The PLA does not refer to our risk assessment tool as the "Safety Monitor." The PLA refers to it as the "On-line Risk Monitor." (See FPL submittal section 3.2.3)

The SE states/implies that we now have a program in place which meets the CRMP requirements, instead of stating that we will implement a program as part of our TS implementation.

50-335/389
2/21/2000

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St Lucie 1 and 2 Request for Revised Implementation Period Unit 1 Amendment 164 and Unit 2 Amendment 105

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Docket: 05000335
Docket: 05000389

FPL

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

February 21, 2000

L-2000-49
10 CFR 50.4

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D C 20555

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Request for Revised Implementation Period
Unit 1 Amendment 164 and Unit 2 Amendment 105

On February 15, 2000, the NRC issued St. Lucie Unit 1 License Amendment 164 and Unit 2 License Amendment 105 with implementation periods of thirty days from the date of issuance. Florida Power & Light Company (FPL) requested, pursuant to Section 2.4.3 of Attachment 1 of NRC Office Letter 803, *Guidance for Processing License Amendments*, a sixty day implementation period for these amendments. As permitted by Office Letter 803, FPL notified the NRC Project Manger and Back-up Project Manager informally, via electronic mail, of the requested implementation period on November 30, 1999. Effective December 1, 1999, the NRC changed the St. Lucie Project Manager and FPL's request was lost in the transition.

Please revise the implementation period for these license amendments to sixty days from the date of receipt. This is the first risk-informed license amendment increasing the allowed outage time. The changes modify the Technical Specifications to extend the allowed outage time (AOT) for a single low pressure safety injection (LPSI) train from 72 hours to 7 days. As part of the amendment approval, FPL committed to implement a Configuration Risk Management Program (CRMP) that puts a procedure-based probabilistic risk assessment process in place that ensures FPL assesses the overall impact of plant maintenance on plant risk. FPL needs the additional time to implement the license amendment to allow time to finalize the CRMP and train the station personnel in its use.

In the future, FPL requests that all license amendments for St. Lucie Units 1 and 2 be issued with a standard implementation period of sixty days from receipt. FPL will notify the NRC if a longer implementation period is required for a specific license amendment. Please contact us if there are any questions.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/GRM

ML003686986

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

A001

50-335/389
2/16/2000

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Subject:
St. Lucie Units 1 and 2 Proposed License Amendment Regarding Accident Monitoring Instrumentation and MOV Thermal Overload Bypass

Body:

Docket: 05000335, Notes: N/A

Docket: 05000389, Notes: N/A

AA2

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

FPL

February 16, 2000

L-2000-002
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Proposed License Amendment
Accident Monitoring Instrumentation
And MOV Thermal Overload Bypass

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 by incorporating the attached Technical Specifications (TS) revisions. The proposed amendments are associated with the motor operated valve thermal overload protection bypass device TS for Unit 2 and the accident monitoring instrumentation TS for both Units 1 and 2.

These changes are requested to correct existing errors in the Technical Specifications. Attachment 1 is an evaluation of the proposed changes. Attachment 2 is the "Determination of No Significant Hazards Consideration." Attachments 3 and 4 contain copies of the affected Technical Specifications pages marked up to show the proposed changes.

The proposed amendments have been reviewed by the St. Lucie Facility Review Group and the FPL Company Nuclear Review Board. In accordance with 10 CFR 50.91 (b) (1), copies of the proposed amendments are being forwarded to the State Designee for the State of Florida. Please contact us if there are any questions about this submittal.

Very truly yours,

Rajiv S. Kundalkar

Rajiv S. Kundalkar
Vice President
St. Lucie Plant

RSK/EJW/KWF

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. W. A. Passetti, Florida Department of Health and Rehabilitative Services

0036854?9

A001

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

Rajiv S. Kundalkar being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Rajiv S. Kundalkar

Rajiv S. Kundalkar

STATE OF FLORIDA

COUNTY OF St. Lucie

Sworn to and subscribed before me

this 16 day of February, ~~19~~ 2000

by Rajiv S. Kundalkar, who is personally known to me.

Leslie J. Whitwell

Signature of Notary Public-State of Florida

Leslie J. Whitwell
MY COMMISSION # CC646183 EXPIRES
May 12, 2001
BONDED THRU TROY FAIN INSURANCE, INC.

Name of Notary Public (Print, Type, or Stamp)

Attachment 1 to FPL Letter L-2000-002

EVALUATION OF PROPOSED TS CHANGES

EVALUATION OF PROPOSED TS CHANGES

Introduction

Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 by incorporating the attached Technical Specifications (TS) revisions. The proposed amendments are associated with:

- the accident monitoring instrumentation TS for both Units 1 and 2,
- the motor operated valve (MOV) thermal overload protection bypass device TS for Unit 2, and
- an administrative change to the Unit 2 TS Index.

Background/Discussion

Unit 1 Accident Monitoring Instrumentation

FPL discovered an apparent discrepancy in the processing of Unit 1 license amendment number 37. Along with other changes based on the NRC's "Category A" TMI related lessons learned recommendations, St. Lucie Unit 1 license amendment number 37 incorporated new Technical Specification requirements for accident monitoring instrumentation (TS 3/4.3.3.8 and Table 3.3-11). ACTION statement 1 states:

"With the number of OPERABLE channels less than required by Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours."

Table 3.3-11 has two columns that describe the number of instrument channels used to monitor specific accident parameters; one column is for the TOTAL NO. OF CHANNELS, and the other column is for the MINIMUM CHANNELS OPERABLE. As the TS is written, it is not clear if the ACTION statement applies to the TOTAL NO. OF CHANNELS or the MINIMUM CHANNELS OPERABLE column.

FPL concluded that an apparent error was made during review, approval, and implementation of the proposed TS changes that were approved by amendment 37. FPL submitted letter L-80-367, dated October 31, 1980, with proposed wording for TS 3.3.3.8 that included two ACTION statements. A 48 hour ACTION Statement 3.3.3.8.a was proposed that applied when the number of operable channels was less than the MINIMUM CHANNELS OPERABLE requirements of Table 3.3-11. A 30 day ACTION Statement 1 for Table 3.3-11 was proposed that applied when the number of operable channels is less than that specified in the TOTAL NO. OF CHANNELS column. As approved, Unit 1 TS amendment 37 incorporated ACTION 1 of Table 3.3-11, but inexplicably omitted the 48 hour ACTION Statement requirement contained in TS 3.3.3.8.a. Unit 1 Technical Specification 3.3.3.8, "Accident Monitoring Instrumentation," is not conservative in that it does not contain a more restrictive ACTION statement when the number of operable channels is less than the minimum channels operable requirements. Additionally, the originally proposed allowed outage time (AOT) for the auxiliary feedwater flow (AFW) instrumentation was more restrictive than the AOT for an out of service AFW pump. This proposed license amendment (PLA) will add appropriate ACTION statements that apply when the number of operable instrument channels is less than the minimum channels operable.

Unit 2 Accident Monitoring Instrumentation

The St. Lucie Unit 2 TS Table 3.3-10, "Accident Monitoring Instrumentation," erroneously indicates the narrow range reactor coolant outlet temperature T_{hot} instrumentation is used to satisfy TS accident monitoring requirements. The TS bases indicates that the intent of this TS table is to provide a post accident instrumentation capability that is consistent with the requirements of Regulatory Guide 1.97. In response to the post TMI NUREGs and Regulatory Guide 1.97, separate wide range reactor coolant outlet temperature T_{hot} instrumentation was installed prior to issuance of the St. Lucie Unit 2 operating license. However, TS Table 3.3-10 was not revised to reflect the use of wide range T_{hot} instrumentation. This PLA will correct the T_{hot} instrumentation error in Table 3.3-10.

Unit 2 Electrical MOV Thermal Overload Protection Bypass Devices

Unit 2 TS Surveillance 4.8.4 requires the periodic surveillance of the thermal overload protection bypass devices associated with the list of MOVs provided in TS Table 3.8-1. This TS table includes valves MV-21-4A and MV-21-4B, which were modified via plant modification (PC/M) 268-292 such that they no longer perform a safety function and are no longer MOVs.

PC/M 268-292 included a 10 CFR 50.59 safety evaluation. This PC/M evaluated the relocation and de-energization of MV-21-4A and 4B and concluded these changes did not represent an Unreviewed Safety Question per 10 CFR 50.59. The PC/M also considered the modification's effect on the Technical Specifications and erroneously concluded that "The subject modification does not affect or impact the Technical Specifications." TS Table 3.8-1 should have been identified as requiring revision to delete MV-21-4A and 4B. This PLA will delete the subject valves from TS Table 3.8-1.

Proposed Changes: Description and Bases/Justification

The affected TS pages, marked up to show the proposed changes, are included in Attachment 3 for Unit 1 and Attachment 4 for Unit 2.

Description and Bases/Justification of Proposed Changes

Unit 1 TS Table 3.3-11

For Unit 1 TS Table 3.3-11, add ACTION 6 which reads as follows:

> "ACTION 6 – With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours."

This action statement will be applicable to the following instrumentation listed in Table 3.3-11: Pressurizer Water Level, RCS Subcooling Margin Monitor, Incore Thermocouples, and Containment Pressure.

The basis and justification for this change is that the addition of the proposed action statement is more conservative than the existing TS and is consistent with the AOT and ACTION statements originally proposed by FPL in letter L-80-367 dated October 31, 1980.

For Unit 1 TS Table 3.3-11, add ACTION 7 that applies to the AFW flow rate instrumentation which reads as follows:

> "ACTION 7 – With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours."

The basis and justification for this change is that the AOT is consistent with the 72 hour AOT for an inoperable AFW pump in accordance with the ACTION statement for TS 3.7.1.2 which reads as follows:

> "With one auxiliary feedwater pump inoperable, restore at least three auxiliary feedwater pumps (two motor driven pumps and one capable of being powered by an OPERABLE steam supply system) to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours."

This change is necessary to ensure that the AOT for AFW flow rate instrumentation is not more conservative than the AOT for the AFW pump itself.

For TS Table 3.3-11, revise ACTION 1 to read as follows:

> "ACTION 1 – With the number of OPERABLE accident monitoring channels less than the Total No. of Channels shown in Table 3.3-11..."

The basis and justification for this change is that it clarifies that ACTION 1 applies when the number of operable channels is less than the Total No. of Channels column in Table 3.3-11.

Unit 2 TS Table 3.3-10

For Unit 2 TS Table 3.3-10 revise item 2 of the INSTRUMENT column to read:

> "Reactor Coolant Outlet Temperature – T_{hot} (Wide Range)"

The basis and justification for this change is that the wide range reactor coolant outlet temperature instrumentation was installed prior to issuance of the St. Lucie Unit 2 operating license in order to meet the requirements of Regulatory Guide 1.97. The design of these instruments is consistent with their use during post accident conditions (e.g., environmental qualification, Class 1E safety related power requirements, etc.). The FPL final response to Regulatory Guide 1.97 (L-85-417) credited the use of the wide range reactor coolant outlet temperature instrumentation to meet Regulatory Guide 1.97 requirements. However, TS Table 3.3-10 was not revised to incorporate this design change made in response to Regulatory Guide 1.97. Therefore, this change is acceptable.

Unit 2 Table 3.8-1

Delete valves MV-21-4A and MV-21-4B.

The bases and justification for this change are as follows. Valves MV-21-4A and MV-21-4B provide isolation of the lube water supply lines to the circulating water (CW) pumps. Prior to the plant modification performed under PC/M 268-292, the CW pump lube water supply consisted of one tap off of each intake cooling water (ICW) essential (safety related) header. As such, MV-21-4A and 4B used to perform a safety function by isolating the non-safety CW pump lube water from the safety related ICW system upon receipt of a safety injection actuation signal (SIAS). As a result of the plant modification, the CW pump lube water supply was relocated to the non-essential header of the ICW system, commonly referred to as the turbine cooling water (TCW) header, and MV-21-4A and 4B were de-energized, effectively becoming manual valves.

The non-safety TCW header is automatically isolated from the safety related ICW headers via MV-21-2 and MV-21-3 (these valves are listed in Table 3.8-1), which automatically close on SIAS. As such, MV-21-4A and 4B no longer perform a safety function and are not required to close upon SIAS. Therefore, this change to Table 3.8-1 is acceptable.

Unit 2 TS Index

When the smooth pages were developed for the Reload Process Improvement proposed license amendment submitted by FPL via letter L-98-308, dated December 18, 1998, the reformatted text carried over onto a new page. Page XIX of the Unit 2 TS was not changed to reflect the new pagination. This change is administrative and required to correct a typographical error in the Unit 2 TS Index.

Environmental Consideration

The proposed license amendment changes requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendment involves no significant hazards consideration and meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and that, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendment.

Conclusion

FPL determined that these proposed license amendments are necessary to correct existing errors in the Technical Specifications.

Attachment 2 to FPL Letter L-2000-002

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Description of amendment request: The proposed license amendments (PLAs) to Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 are necessary to:

- Correct an existing non-conservatism in Unit 1 Technical Specification (TS) 3.3.3.8, Accident Monitoring Instrumentation, by providing a 48 hour action statement that applies when the number of operable accident monitoring instrumentation channels is less than the minimum channels operable requirements. This is consistent with FPL's original submittal as documented in FPL letter L-80-367 dated October 31, 1980.

- Add a 72 hour ACTION statement to Unit 1 Technical Specification (TS) 3.3.3.8, Accident Monitoring Instrumentation, that applies when the number of operable auxiliary feedwater flow instrumentation channels is less than the minimum channels operable requirements.

- Correct an existing error in Unit 2 Technical Specification (TS) Table 3.3-10, Accident Monitoring Instrumentation, that incorrectly indicates the use of reactor coolant outlet temperature T_{hot} narrow range instrumentation to meet Regulatory Guide 1.97 accident monitoring instrumentation requirements. St. Lucie Unit 2 utilizes wide range reactor coolant outlet temperature T_{hot} instrumentation to meet Regulatory Guide 1.97 accident monitoring requirements.

- Correct an existing error in Unit 2 TS Table 3.8-1, Motor Operated Valves Thermal Overload Protection Bypass Devices, by deleting valves MV-21-4A and MV-21-4B from the list. These valves were changed to manual valves and no longer perform a safety function.

- Correct an administrative error in the Unit 2 TS Index.

Pursuant to 10 CFR 50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows.

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The addition of the new ACTION statements for the Unit 1 accident monitoring instrumentation adds conservatism that does not exist in the current Technical Specifications. These changes are consistent with either FPL's originally proposed license amendment for this instrumentation or consistent with the Technical Specification allowed outage time for the component being monitored (i.e., the auxiliary feedwater pumps). Unit 2 valves MV-21-4A and MV-21-4B were modified to be manually operated valves and no longer perform an accident mitigation function. Unit 2 wide range T_{hot} instrumentation is used to satisfy Regulatory Guide 1.97 accident monitoring requirements.

These Technical Specification changes either correct existing errors or add conservatism to the way the Unit is operated. Based on the above, the physical changes to plant equipment or plant operation would not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

Accident monitoring instrumentation monitors the process of postulated events, and is not an accident initiator. Unit 2 valves MV-21-4A and MV-21-4B were modified to be manually operated valves and no longer have an active safety function, therefore, these valves are not accident initiators. These Technical Specification changes either correct existing errors or add conservatism to the way the Unit is operated. Based on the above, the physical changes to plant equipment or plant operation would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The proposed amendments do not involve a significant reduction in a margin of safety. FPL determined that these proposed license amendments are necessary to correct existing errors or add conservatism to the way the Unit is operated. As such, the assumptions and conclusions of the accident analyses in the UFSAR remain valid and the associated safety limits will continue to be met.

Based on the above discussion and the supporting Evaluation of Technical Specification changes, FPL has determined that the proposed license amendments involve no significant hazards consideration.

Attachment 3 to FPL Letter L-2000-002

ST. LUCIE UNIT 1 MARKED UP TECHNICAL SPECIFICATION PAGES

Page 3/4 3-42

Page 3/4 3-43

TABLE .3-11

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	TOTAL NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE	ACTION
1. Pressurizer Water Level	3	1	1,6
2. Auxiliary Feedwater Flow Rate	1/pump	1/pump	~~1~~ 7
3. RCS Subcooling Margin Monitor	2	1	1,6
4. PORV Position Indicator Acoustic Flow Monitor	1/valve	1/valve	2
5. PORV Block Valve Position Indicator	1/valve	1/valve	2
6. Safety Valve Position Indicator	1/valve	1/valve	3
7. Incore thermocouples	4/core quadrant	2/core quadrant	1,6
8. Containment Sump Water Level (Narrow Range)	1*	1*	4, 5
9. Containment Sump Water Level (Wide Range)	2	1	4, 5
10. Reactor Vessel Level Monitoring System	2**	1**	4, 5
11. Containment Pressure	2	1	1,6

*The non-safety grade containment sump water level instrument may be substituted.

**<u>Definition of OPERABLE</u>: A channel is composed of eight (8) sensors in a probe, of which four (4) sensors must be OPERABLE.

TABLE 3.3-11 (Continued)

ACTION STATEMENTS

the Total No. of Channels shown

ACTION 1 - With the number of OPERABLE channels less than ~~required by~~ in Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 30 days or be in HOT STANDBY within the next 12 hours.

ACTION 2 - With position indication inoperable, restore the inoperable indicator to OPERABLE status or close the associated PORV block valve and remove power from its operator within 48 hours or be in HOT STANDBY within the next 6 hours.

ACTION 3 - With any individual valve position indicator inoperable, obtain quench tank temperature, level and pressure information once per shift to determine valve position.

ACTION 4 - With the number of OPERABLE Channels one less than the Total Number of Channels shown in Table 3.3-11, either restore the inoperable channel to OPERABLE status within 7 days if repairs are feasible without shutting down or prepare and submit a Special Report to the Commission pursuant to the specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 5 - With the number of OPERABLE Channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours if repairs are feasible without shutting down or:

1. Initiate an alternate method of monitoring the reactor vessel inventory; and

2. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status; and

3. Restore the Channel to OPERABLE status at the next scheduled refueling.

Insert 1

Insert 1:

ACTION 6 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 48 hours or be at least in HOT SHUTDOWN within the next 12 hours.

ACTION 7 - With the number of OPERABLE accident monitoring channels less than the Minimum Channels OPERABLE requirements of Table 3.3-11, either restore the inoperable channel(s) to OPERABLE status within 72 hours or be at least in HOT SHUTDOWN within the next 12 hours.

Attachment 4 to FPL Letter L-2000-002

ST. LUCIE UNIT 2 MARKED UP TECHNICAL SPECIFICATION PAGES

Page XIX

Page 3/4 3-42

Page 3/4 8-19

TABLE 3.. J

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT		REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE
1.	Containment Pressure	2	1
2.	Reactor Coolant Outlet Temperature - T_{Hot} ~~(Narrow Range)~~ (Wide Range)	2	1
3.	Reactor Coolant Inlet Temperature - T_{Cold} (Wide Range)	2	1
4.	Reactor Coolant Pressure - Wide Range	2	1
5.	Pressurizer Water Level	2	1
6.	Steam Generator Pressure	2/steam generator	1/steam generator
7.	Steam Generator Water Level - Narrow Range	1/steam generator	1/steam generator
8.	Steam Generator Water Level - Wide Range	1/steam generator*	1/steam generator*
9.	Refueling Water Storage Tank Water Level	2	1
10.	Auxiliary Feedwater Flow Rate (Each pump)	1/pump*	1/pump*
11.	Reactor Cooling System Subcooling Margin Monitor	2	1
12.	PORV Position/Flow Indicator	2/valve***	1/valve**
13.	PORV Block Valve Position Indicator	1/valve**	1/valve**
14.	Safety Valve Position/Flow Indicator	1/valve***	1/valve***
15.	Containment Sump Water Level (Narrow Range)	1****	1****
16.	Containment Water Level (Wide Range)	2	1
17.	Incore Thermocouples	4/core quadrant	2/core quadrant
18.	Reactor Vessel Level Monitoring System	2*****	1*****

* These corresponding instruments may be substituted for each other.

** Not required if the PORV block valve is shut and power is removed from the operator.

*** If not available, monitor the quench tank pressure, level and temperature, and each safety valve/PORV discharge piping temperature at least once every 12 hours.

**** The non-safety grade containment sump water level instrument may be substituted.

***** Definition of OPERABLE: A channel consists of eight (8) sensors in a probe of which four (4) sensors must be OPERABLE.

TABLE 3.8.1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION BYPASS DEVICES

VALVE NUMBER		FUNCTION	BYPASS (YES/NO)
MAIN STEAM:			
	MV-08-1A,1B	MSIV BYPASS	YES
	MV-08-18A,18B	A.D.V.	YES
	MV-08-19A,19B	A.D.V.	YES
	MV-08-12,13	AFW TURBINE INLET	YES
	MV-08-3	AFW TURBINE INLET	YES
	MV-08-14,15,16,17	A.D.V. ISOL.	YES
MAIN FEEDWATER:			
	MV-09-9,10,11,12	AUX. FEED ISOL.	YES
	MV-09-13,14	AUX. FEED X-TIE	YES
ICW:	MV-21-2,3	ICW ISOL.	YES
	~~MV-21-4A,4B~~	~~ICW ISOL.~~	~~YES~~
CCW:	MV-14-17,18,19,20	FUEL POOL ISOL.	YES
	MV-14-9,10,11,12,13,14,15, 16	CONT. FAN ISOL.	YES
	MV-14-1,2,3,4	CCW PUMP ISOL.	YES
C.S.:	MV-07-1A,1B	RWT ISOL.	YES
	MV-07-2A,2B	SUMP ISOL.	YES
	MV-07-3,4	SYSTEM ISOL.	YES
HVAC:	FCV-25-14,15,16,17,18,19	CRECS ISOL.	YES
	FCV-25-24,25	CRECS ISOL.	YES
	FCV-25-11,12	SBVS ISOL.	YES
	FCV-25-35	VENT ISOL.	YES
	FCV-25-29,34	H2 CONT. PURGE	YES
	FCV-25-30,31	SFP EXHAUST	YES
	FCV-25-32,33	SBVS INLET	YES

1/19/2000

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Body:

Docket: 05000335, Notes: N/A

Docket: 05000389, Notes: N/A

AA2

FPL

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

L-2000-001
10 CFR 50.90

January 19, 2000

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

RE: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Proposed License Amendments
Main Steam and Pressurizer Code Safety Valve
Setpoint Setting and Setpoint Testing Requirements

Pursuant to 10 CFR 50.90, Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 by incorporating the attached Technical Specifications (TS) revisions. These proposed license amendments will revise the Unit 1 and 2 Technical Specifications to be consistent with the Standard Technical Specifications (STS) requirements that allow for an expanded as-found testing acceptance tolerance (i.e., beyond ±1%) for the main steam safety valves (MSSVs) and pressurizer code safety valves (PSVs), whereas the existing St. Lucie Technical Specifications do not. Expanding the as-found acceptance limits will allow the test program to accept MSSVs and PSVs whose setpoints are found to be within accident analysis assumptions. The ±1% as-left criteria will remain unchanged. Mode 5 operability requirements for the PSVs will also be deleted.

Attachment 1 is an evaluation of the proposed changes. Attachment 2 is the "Determination of No Significant Hazards Consideration." Attachments 3 and 4 contain copies of the affected Technical Specifications pages marked up to show the proposed changes.

The proposed amendments have been reviewed by the St. Lucie Facility Review Group and the FPL Company Nuclear Review Board. In accordance with 10 CFR 50.91 (b) (1), copies of the proposed amendments are being forwarded to the State Designee for the State of Florida. Please contact us if there are any questions about this submittal.

Very truly yours,

J. A. Stall
Vice President
St. Lucie Plant

JAS/EJW/KWF

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. W. A. Passetti, Florida Department of Health and Rehabilitative Services

A001

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

J. A. Stall being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

J. A. Stall

STATE OF FLORIDA

COUNTY OF St. Lucie

Sworn to and subscribed before me

this 19 day of January, ~~19~~ 2000

by J. A. Stall, who is personally known to me.

Leslie J. Whitwell

Signature of Notary Public-State of Florida

Leslie J. Whitwell
MY COMMISSION # CC646183 EXPIRES
May 12, 2001
BONDED THRU TROY FAIN INSURANCE, INC.

Leslie J. Whitwell
Name of Notary Public (Print, Type, or Stamp)

Attachment 1 to FPL Letter L-2000-001

EVALUATION OF PROPOSED TS CHANGES

Introduction

Florida Power and Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 by incorporating the attached Technical Specifications (TS) revisions. These proposed license amendments (PLA) will revise the Unit 1 and 2 main steam safety valve (MSSV) and pressurizer code safety valve (PSV) Technical Specifications to be consistent with the Standard Technical Specifications (STS). The STS include a provision for MSSV and PSV setpoint setting and setpoint testing that currently does not exist in the St. Lucie Technical Specifications. Specifically, the STS allow an expanded setpoint surveillance testing acceptance tolerance (i.e., beyond ±1%), whereas the existing St. Lucie Technical Specifications do not. Expanding the as-found acceptance limits will allow the test program to accept MSSVs and PSVs whose setpoints are found to be within accident analysis assumptions. The ±1% as-left criteria will remain unchanged.

St. Lucie Plant and industry experience with safety valves has shown that valve setpoint drift is a common phenomena whereby, over time, the as-found setting of a safety valve may "drift" beyond the original setting tolerance of ±1%. Both the ASME Code and the Standard Technical Specifications for Combustion Engineering Plants recognize setpoint drift and both allow for expanded as-found acceptance limits. The ASME Code accepts a deviation of up to +3% (the Code does not address a negative tolerance). The Standard Technical Specifications, based in part on the ASME Code, accepts a deviation of up to ±3%.

These PLAs will also revise the applicability of the PSV LCOs to be consistent with the STS. Currently, there is a PSV LCO for Modes 1, 2, and 3 and a second PSV LCO for Modes 4 and 5. The STS format consists of a single PSV LCO, applicable in Modes 1, 2, 3 and Mode 4 with RCS cold leg temperatures above the low temperature overpressure protection (LTOP) limit; below the LTOP temperature limit a separate LTOP LCO is applicable.

Background/Discussion

The setpoint tolerance for the MSSVs and the PSVs is currently ±1%. This ±1% band has been used as an acceptance criterion for the periodic lift testing of the valves. Per Technical Specification Surveillance Requirements, MSSVs and PSVs are lift tested in accordance with the Inservice Testing Program, procedure ADM-29.01, "Inservice Testing (IST) Program for Pumps and Valves." This lift testing has historically produced results where one or more of the tested valves lifts at a pressure that is outside the ±1% band. Although the as-found lift pressure may be bounded by the accident analysis, FPL is required to initiate a licensee event report (LER) as a condition prohibited by the Technical Specifications because they do not explicitly allow for an as-found lift setpoint tolerance.

St. Lucie Plant and industry experience with safety valves has shown that valve setpoint drift is a common phenomena whereby, over time, the as-found setting of a safety valve may "drift" beyond the original setting tolerance of ±1%. Both the ASME Code, ASME/ANSI OM-1987, "Operation and Maintenance of Nuclear Power Plants, Part 1 – Requirements for Inservice Performance Testing of Nuclear Power Plant Pressure Relief Devices," and the Standard Technical Specifications for Combustion Engineering Plants, NUREG-1432, recognize setpoint drift and both allow for expanded as-found acceptance limits. The ASME Code accepts a deviation of up to +3% (the Code does not address a negative tolerance). The Standard Technical Specifications accepts a deviation of up to ±3%.

Description of the Proposed Changes

The proposed Technical Specification changes are summarized below. Marked-up Technical Specification pages for this proposed change are provided as Attachments 3 and 4.

Note - This evaluation does not revise the requirement to set the MSSVs and PSVs within a tolerance of ±1% .

MSSVs

Technical Specification 3.7.1.1 (both units), Turbine Cycle Safety Valves, requires the MSSVs to be operable with settings in accordance with Table 4.7-1 (Unit 1) and Table 3.7-2 (Unit 2). Both of these tables identify a setpoint tolerance of ±1%. The Bases for Technical Specification 3.7.1.1 (both units) do not discuss setpoint tolerances and/or setpoint drift.

Section 3.7.1 of the STS provides a clear distinction between the tolerance used for setting of the lift setpoint and the tolerance used for surveillance testing of the lift setpoint. Specifically, Surveillance Requirement 3.7.1.1 states the following:

> Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%.

STS Table 3.7.1-2 lists the MSSVs and identifies the lift setting as "psig ±[3]%" (the use of the "[]" parentheses indicates that a plant specific value is to be provided). The Bases for this STS specification states:

> Table 3.7.1-2 allows a ±[3%] setpoint tolerance for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift.

The proposed changes of Attachments 3 and 4 provide an expanded setpoint tolerance for valve testing that is based on existing plant accident analyses. The Bases of the above MSSV Technical Specifications will be revised to distinguish between the as-found surveillance testing acceptance criteria added by this evaluation and the unchanged as-left setpoint acceptance criteria. Table 4.7-1 (Unit 1) and Table 3.7-2 (Unit 2) will be revised to reflect the as-left setpoint range rather than a specific setpoint value. Although this table format is slightly different than the table format presented in the STS, it will provide consistency with the format of the PSV specification (the STS formats for the PSV and MSSV specifications are inconsistent). The wording of the Action for TS 3.7.1.1 is being revised to change the "Cold Shutdown...30 hours" requirement to a "Hot Shutdown...12 hours" requirement. This is consistent with the STS and is acceptable since the LCO does not require MSSV operability in Hot Shutdown. Action "b" is retained in lieu of using the STS note regarding Modes 1 and 2 applicability of the Surveillance Requirement.

In addition to the above changes, the MSSV orifice size is being deleted from Table 4.7-1 (Unit 1) and Table 3.7-2 (Unit 2). The orifice size is a design feature that cannot be changed without a formal modification to the valves. The STS do not include valve orifice size requirements.

PSVs

Technical Specifications 3.4.2 (Unit 1) and 3.4.2.1 (Unit 2), Reactor Coolant System Safety Valves – Shutdown, and Technical Specifications 3.4.3 (Unit 1) and 3.4.2.2 (Unit 2), Safety Valves – Operating, require the PSVs to be operable with a lift setting of 2500 psia ±1%. The Bases for these Technical Specifications do not discuss setpoint tolerances and/or setpoint drift.

Section 3.4.10 of the STS provides a clear distinction between the tolerance used for setting of the lift setpoint and the tolerance used for surveillance testing of the lift setpoint. Specifically, Surveillance Requirement 3.4.10.1 states the following:

> Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ±1%.

The Bases for this STS specification states:

> The pressurizer safety valve setpoint is ±[3%] for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift.

The proposed changes of Attachments 3 and 4 provide an expanded setpoint tolerance for valve testing that is based on existing plant accident analyses. As such, the referenced Technical Specifications will be revised to note these setpoint tolerances. Also, the wording of the associated LCOs, Actions, Surveillance Requirements, and corresponding Bases will be revised in a manner similar to the STS.

Note that the STS have a single PSV LCO that is applicable in Modes 1, 2, 3 and Mode 4 >[285]°F. At or below [285]°F a separate LTOP LCO is applicable and the PSVs are no longer required to be operable. The St. Lucie TS currently include two LCOs for RCS overpressure protection in Modes 4 and 5: one LCO for the PSVs and one LCO for the LTOP system. The St. Lucie TS also include an LCO for PSV operability in Modes 1, 2, and 3. These proposed license amendments revise the St. Lucie TS to adopt the STS format. This will delete the "overlap" in required RCS overpressure protection.

In addition, the LCOs will be revised to reflect the as-left PSV setpoint acceptance tolerance values in units of psig in lieu of psia. The actual setpoint value of 2500 psia is not being changed. The acceptance tolerance range is calculated based on the setpoint value of 2500 psia, then converted to units of psig. The Bases will be revised to explain the above. This change will make the Technical Specifications consistent with implementing procedures, which use units of psig.

Bases/Justification of the Proposed Changes

UFSAR

Section 5.2.2.1 of the Unit 1 UFSAR and Section 10.3.2 of the Unit 2 Updated Final Safety Analysis Report (UFSAR) provide similar descriptions of the MSSVs. The MSSVs are described as providing overpressure protection for the shell side of the steam generators and the main steam line piping up to the main steam isolation valves. The MSSVs are ASME Code, spring loaded, open bonnet, flange mounted safety valves that discharge to atmosphere. There are eight MSSVs installed on each main steam header, four of which are set at 985.3 psig (1000 psia) and four of which are set at 1025.3 psig (1040 psia). Table 5.5-2 (Unit 1) and Table 10.3-1 (Unit 2) provide design information on the MSSVs, including setpoint values. There is no detailed discussion in the UFSARs regarding setpoint testing requirements and/or acceptance limits.

Section 5.5.3 of the Unit 1 UFSAR and Section 5.4.13 of the Unit 2 UFSAR provide similar descriptions of the PSVs. The PSVs are described as providing overpressure protection for the reactor coolant system. The PSVs are ASME Code, spring loaded, enclosed bonnet, flange mounted safety valves that discharge to the quench tank. There are three PSVs installed on top of the pressurizer, each of which is set at 2485.3 psig (2500 psia). Table 5.5-4 (Unit 1) and Table 5.4-8 (Unit 2) provide design information on the PSVs, including setpoint values. There is no detailed discussion in the UFSARs regarding setpoint testing requirements and/or acceptance limits.

For both the MSSVs and the PSVs, Unit 1 and 2 UFSAR Chapter 15 accident analyses describe valve actuation assumed at varying pressure values. The values assumed therein are for analysis purposes only and are not intended to represent setpoint requirements. A review of setpoint tolerances with respect to plant accident analyses is provided in the evaluation below.

Evaluation and Justification of Changes

The Reference 3 Standard Technical Specifications distinguish between the safety valve setpoint tolerance and the tolerance used for surveillance testing of the MSSVs and PSVs. This distinction is made in the Bases section of the STS. Specifically, the STS describe a "±[3%] setpoint tolerance for OPERABILITY; however, the valves are reset to ±1% during the Surveillance to allow for drift." Note that the STS refer to testing in accordance with the ASME Code and the ±3% value is likely used in the STS since +3% is the value recognized in the Code (Reference 2) as a tolerance beyond which the valve is considered to have failed its lift test. The Code does not address negative tolerances.

As noted above, MSSV and PSV setpoints are established within a ±1% tolerance; however, as identified in the STS, plant specific tolerances should be provided for determination of operability. The tolerance to be specified for valve operability is logically limited by the ability of each plant's accident analyses to accommodate the tolerance. FPL reviewed the relevant accident analyses to provide the following analysis.

Setpoint Tolerances - St. Lucie Unit 1

The safety analysis inputs are documented in SAPP-1, Revision 3, "St. Lucie Unit 1 Safety Analysis Plant Parameters." The safety analysis impacted by the PSV and the MSSV set pressures are:

1) Loss of External Load (LOEL), and
2) Small Break LOCA

The analysis for LOEL assumes that openings of safety valves do not occur until the pressure reaches a value corresponding to valve tolerance of +1% for MSSVs and +3% for PSVs (SPC report EMF-96-135, "St. Lucie Unit 1 Chapter 15 Event Review and Analysis for 30% Steam Generator Tube Plugging," May, 1996 and SPC letter TMH:97:210, T.M. Howe (SPC) to J. Polavarapu (FPL), "Assessment of Change in Calculated Peak Secondary Side Pressure due to Changes in Modeling of St. Lucie Unit 1 MSSVs," July 16, 1997). The negative tolerance for the MSSVs has no adverse effect on this analysis.

For the PSVs, the tolerance on the negative side is limited by the reactor trip, such that the PSVs do not open prior to the trip setpoint. The analysis assumes late opening of the PSVs to maximize the overpressure calculation. The high pressure trip setpoint is 2400 psia with an uncertainty of 22 psi. Since the valve lift setpoint is 2500 psia, a tolerance on the negative side of up to 2.5% will not affect the analysis results, assuming a conservative uncertainty of 37 psi on the trip setpoint.

The analysis for a small break LOCA supports an MSSV setpoint tolerance of +3% (SPC report EMF-1987, "St. Lucie Unit 1 Small Break LOCA Analysis with Asymmetric HPSI Flow," November, 1997 and SPC letter RIW:99:141, R.I. Wescott (SPC) to R.J. Rodriguez (FPL), "Transmittal of Small Break LOCA FSAR Update Package," June 14, 1999). Since

there is no analytical concern regarding the negative setpoint tolerance for the MSSVs, the STS value of –3% is used.

The following as-found setpoint tolerance limits are therefore acceptable based on the current UFSAR analysis:

MSSVs	-	+1%, -3%
PSVs	-	+3%, -2.5%

Setpoint Tolerances - St. Lucie Unit 2

The analysis inputs for St. Lucie Unit 2 are documented in SAPP-2, Revision 4, "St. Lucie Unit 2 Safety Analysis Plant Parameters," and the values supported by the analysis, submitted to the NRC as part of Reload Process Improvement (RPI), are presented in ABB letter F2-99-048, G. Singh (ABB-CE) to R. J. Rodriguez (FPL), "Transmittal of St. Lucie Unit 2 Reload Checklist Document, Revision 3," May 24, 1999. Since there is no analytical concern regarding the negative setpoint tolerance for the MSSVs, the STS value of –3% is used. Based on these references, the following values of MSSV and PSV as-found setpoint tolerances are supported by the UFSAR analysis to be incorporated in Cycle 12:

MSSVs	-	+1%, -3%
PSVs	-	±2%

Since the above analysis was performed for Cycle 12 and the plant is currently operating in Cycle 11, a review was performed to determine the applicability of the proposed tolerances for EOC 11 setpoint testing. The review concluded the RPI analysis bounds the conditions at EOC 11; therefore, the setpoint tolerances proposed above remain applicable for the year 2000 refueling outage.

Note that for both the Unit 1 and Unit 2 analyses above, the setpoint tolerances were calculated using the current Technical Specification MSSV and PSV setpoint values in units of psia.

Deletion of MSSV Orifice Size

The MSSV orifice size data is being deleted from Table 4.7-1 (Unit 1) and Table 3.7-2 (Unit 2). While it is appropriate for the Technical Specifications to identify safety valve setpoints and to provide surveillance testing to ensure proper valve performance, safety valve orifice size is a passive, fixed design feature that cannot "drift" or otherwise be changed without a formal modification to the valves. There is no existing TS Surveillance associated with this design feature and the STS do not include valve orifice size requirements. As such, it is acceptable to delete this unnecessary detail from the Technical Specifications.

PSV LCO Reformatting – Unit 1

The original Unit 1 TS, issued in 1976, did not include an LCO specifically for LTOP conditions. RCS overpressure protection was provided solely via TS 3.4.2, for PSV operability in Modes 4 and 5, and TS 3.4.3, for PSV operability in Modes 1, 2, and 3. TS Amendment #60, issued in 1983, subsequently added TS 3.4.13 for LTOP, which utilizes the PORVs for RCS overpressure protection whenever cold leg temperatures are below a predetermined limit, which includes operation in Modes 4 (partial), Mode 5 and Mode 6. Amendment #60 did not modify the existing PSV LCOs, thus it created an overlap in the TS required RCS overpressure protection provided by the PSVs and PORVs. At the time Amendment #60 was issued it was apparently not recognized that the PSV LCOs should have been revised.

The STS format provides a single LCO for PSV operability and a single LCO for LTOP protection. STS 3.4.10 requires the PSVs to be operable in "MODES 1, 2, and 3, MODE 4 with all RCS cold leg temperatures > [285] °F. STS 3.4.12 requires the PORVs to be operable for LTOP protection in "MODE 4 when any RCS cold leg temperature is ≤ [285] °F, MODE 5, MODE 6..." The Bases for these specifications describe "[285] °F" as the RCS cold leg temperature for which, based on plant-specific analysis, LTOP protection is required. The applicability of these two LCOs is such that they provide for continual RCS overpressure protection from Mode 1 through Mode 6 without any overlap between them.

The proposed changes provided in Attachment 3 will combine TS 3.4.2 and 3.4.3 into a single LCO and will eliminate the PSV LCO applicability for Mode 5. Additionally, PSV Mode 4 applicability will be limited to the condition when all RCS cold leg temperatures are greater than 281°F, which is the point at which, during a cooldown, the PORVs are required to be operable for LTOP protection (TS 3.4.13). The cooldown LTOP "ceiling" of 281°F was selected since it is more limiting than the heatup "ceiling" of 304 °F.

This change is acceptable since it does not diminish the ability of the Technical Specifications to provide required overpressure protection of the RCS and since it is consistent with the STS format. There is no accident analysis that is affected by this change.

PSV LCO Reformatting – Unit 2

With respect to the PSVs and PORVs, the original Unit 2 TS (issued in 1983) were similar to the Unit 1 TS as of Amendment #60. Unit 2 TS 3.4.2.1 requires PSV operability in Modes 4 and 5, and TS 3.4.2.2 requires PSV operability in Modes 1, 2 and 3. Additionally, TS 3.4.9.3 requires the PORVs to be operable for RCS overpressure protection whenever cold leg temperatures are below a predetermined limit, which includes operation in Modes 4 (partial), Mode 5, and Mode 6.

The proposed changes to the Unit 2 TS are similar to those described above for Unit 1 and are provided in Attachment 4. The only significant difference between the units is the temperature for which LTOP is required - 230°F for Unit 2. The justification for this Unit 2 change is the same as for the Unit 1 change.

Environmental Consideration

The proposed license amendments change requirements with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The proposed amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendments involve no significant hazards consideration and meets the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and that, pursuant to 10 CFR 51.22(b), an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendments.

Conclusion

There is no safety significance associated with the proposed changes. The MSSVs and PSVs will continue to be set to the as-left tolerance of ±1% and will perform as assumed in plant accident analyses. The proposed changes simply clarify that as long as MSSV and PSV as-found lift setpoints are within accident analyses assumptions, the valves may be considered operable. This change allows for a limited amount of setpoint drift and is consistent with the provisions of the Standard Technical Specifications and ASME Code. The PSV lower mode LCO changes will provide a clear distinction between PSV and PORV/LTOP requirements while ensuring the required RCS overpressure protection is provided.

Converting the PSV setpoint value to units of psig is acceptable since the actual valve lift setpoint and as-left setpoint tolerances are effectively not changing. Applying the as-found acceptance tolerances to the lift setpoint in units of psia and then converting to units of psig is acceptable since it is consistent with the above accident analysis review.

The proposed Technical Specification changes governing MSSV and PSV as-found set pressures are acceptable since they are based on existing accident analyses assumptions and they are consistent with the STS and ASME Code, which allow up to a ±3% tolerance for as-found setpoint testing. The valves will continue to be set within the existing required ±1% setpoint tolerance.

Attachment 2 to FPL Letter L-2000-001

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Description of amendment request: The proposed license amendments (PLAs) to Facility Operating Licenses DPR-67 for St. Lucie Unit 1 and NPF-16 for St. Lucie Unit 2 will revise the Unit 1 and 2 Technical Specifications to be consistent with the Standard Technical Specifications (STS) requirements for main steam safety valve (MSSV) and pressurizer code safety valve (PSV) setpoint setting and setpoint testing. Specifically, the STS allow an expanded setpoint surveillance testing acceptance tolerance (i.e., beyond ±1%), whereas the existing St. Lucie Technical Specifications do not. There is no safety significance associated with the proposed changes.

Expanding the as-found acceptance limits will allow the test program to accept MSSVs and PSVs whose setpoints are found to be within accident analysis assumptions. The ±1% as-left criteria will remain unchanged. The proposed changes simply clarify that as long as MSSV and PSV as-found lift setpoints are within accident analyses assumptions, the valves may be considered operable. This change allows for a limited amount of setpoint drift and is consistent with the provisions of the Standard Technical Specifications and ASME Code. The PSV lower mode LCO changes will provide a clear distinction between PSV and PORV/LTOP requirements while ensuring the required RCS overpressure protection is provided.

Pursuant to 10 CFR 50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows.

(1) Operation of the facility in accordance with the proposed amendments would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The probability of occurrence of an accident previously evaluated has not been increased. The changes provided in this safety evaluation do not affect the assumptions or results of any accident evaluated in the UFSAR. The actual setpoints and as-left setpoint tolerances of the MSSVs and PSVs are not changed as a result of this evaluation.

Likewise, the consequences of any accident previously evaluated have not been increased. The ability of the MSSVs and PSVs to respond to accident conditions as assumed in any accident analysis has not been affected (i.e., adequate overpressure protection is provided). The proposed changes allow for the acceptance of safety

valve lift test results based on tolerances that are consistent with accident analysis assumptions.

(2) Operation of the facility in accordance with the proposed amendments would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed activity does not create the possibility of an accident of a different type than any previously evaluated. No physical plant changes are being made and no new failure modes have been introduced by the proposed changes. This evaluation revises the acceptance criteria for MSSV and PSV lift test results based on tolerances that are consistent with accident analysis assumptions. The actual setpoints and as-left setpoint tolerances of the MSSVs and PSVs are not changed as a result of this evaluation.

(3) Operation of the facility in accordance with the proposed amendments would not involve a significant reduction in a margin of safety.

The margin of safety as defined in the basis for any Technical Specification or in any licensing document has not been reduced. MSSV and PSV setpoint values are not being changed. MSSV and PSV setpoints are still required to be set within a tolerance of ±1% (the as-left setpoint tolerance). This evaluation allows for the revision of acceptance criteria for MSSV and PSV lift test results such that testing criteria is consistent with accident analysis assumptions. This will allow for the accommodation of setpoint drift without invalidating the accident analyses. The proposed changes are consistent with the Standard Technical Specifications, which require MSSV and PSV setting within a ±1% tolerance, but allow surveillance testing to accept valves that lift within "±[3]%." A review of the plants' accident analyses has identified the plant-specific tolerances that may be used for this surveillance testing. These values have been used in the Attachment 3 and 4 proposed changes.

Conclusion

Based on the above discussion and the supporting Evaluation of Technical Specification changes, FPL has determined that the proposed license amendments involve no significant hazards consideration.

Attachment 3 to FPL Letter L-2000-001

ST. LUCIE UNIT 1 MARKED UP TECHNICAL SPECIFICATION PAGES

TS Page IV

TS Page 3/4 4-2

TS Page 3/4 4-3

TS Page B-3/4 4-1

TS Page B-3/4 4-2

TS Page 3/4 7-1

TS Page 3/4 7-3

TS Page B-3/4 7-2

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

REACTOR COOLANT SYSTEM

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 ± 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.

Deleted

SURVEILLANCE REQUIREMENTS

4.4.2 No additional Surveillance Requirements other than those required by the Inservice Testing Program.

SAFETY VALVES - OPERATING

LIMITING CONDITION FOR OPERATION

3.4.3 All pressurizer code safety valves shall be OPERABLE with ~~a~~ as-left lift settings ~~of 2500 PSIA ±1%.~~ ≥ 2460.3 psig and ≤ 2510.3 psig

APPLICABILITY: MODES ~~1, 2 and 3.~~ 1,2,3, and 4 with all RCS cold leg temperatures > 281 °F

ACTION:

~~With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT SHUTDOWN within 12 hours.~~

Insert 1

SURVEILLANCE REQUIREMENTS

4.4.3 ~~No additional Surveillance Requirements other than those required by the Inservice Testing Program.~~

Insert 2

[illegible]

[illegible]

[illegible]

[illegible]

[illegible]

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 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 restricting starting of the Reactor Coolant Pumps to when the secondary water temperature of each steam generator is less than 30°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 2×10^5 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 the Inservice Testing Program.~~

Insert 3

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 pressuzier heater capacity is capable of maintaining natural circulation sub-cooling. 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 damage or progressive degradation due to design, manufacturing errors, or in-service 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.

3/4.7 PLANT SYSTEMS

3.4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION:

3.7.1.1 All main steam line code safety valves shall be OPERABLE ~~with lift settings~~ as specified in Table 4.7-1.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With ~~both reactor coolant loops and associated steam generators in operation and with~~ one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided that within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 ~~No additional Surveillance Requirements other than those required by the Inservice Testing Program.~~

Insert 4

TABLE 4.7-1

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER			AS-LEFT LIFT SETTING ~~(± 1%)~~	~~ORIFICE SIZE~~
	Header A	Header B		
a.	8201	8205	~~1000 psia~~ ≥975.3 psig and ≤995.3 psig	~~16 in.^{2}~~
b.	8202	8206	~~1000 psia~~ "	~~16 in.^{2}~~
c.	8203	8207	~~1000 psia~~ "	~~16 in.^{2}~~
d.	8204	8208	~~1000 psia~~ "	~~16 in.^{2}~~
e.	8209	8213	~~1040 psia~~ ≥1014.9 psig and ≤1035.7 psig	~~16 in.^{2}~~
f.	8210	8214	~~1040 psia~~ "	~~16 in.^{2}~~
g.	8211	8215	~~1040 psia~~ "	~~16 in.^{2}~~
h.	8212	8216	~~1040 psia~~ "	~~16 in.^{2}~~

PLANT SYSTEMS

BASES

106.5 = Power Level-High Trip Setpoint for two loop operation

X = Total relieving capacity of all safety valves per steam line in lbs/hour (6.192×10^6 lbs/hr.)

Y = Maximum relieving capacity of any one safety valve in lbs/hour (7.74×10^5 lbs/hr.)

Insert 5

3/4.7.1.2 AUXILIARY FEEDWATER PUMPS

The OPERABILITY of the auxiliary feedwater pumps ensures that the Reactor Coolant System can be cooled down to less than 325°F from normal operating conditions in the event of a total loss of off-site power.

Any two of the three auxiliary feedwater pumps have the required capacity to provide sufficient feedwater flow to remove reactor decay heat and reduce the RCS temperature to 325°F where the shutdown cooling system may be placed into operation for continued cooldown.

3/4.7.1.3 CONDENSATE STORAGE TANK

The OPERABILITY of the condensate storage tank with the minimum water volume ensures that sufficient water is available for cooldown of the Reactor Coolant System to less than 325°F in the event of a total loss of off-site power. The minimum water volume is sufficient to maintain the RCS at HOT STANDBY conditions for 8 hours with steam discharge to atmosphere.

3/4.7.1.4 ACTIVITY

The limitations on secondary system specific activity ensure that the resultant off-site radiation dose will be limited to a small fraction of 10 CFR Part 100 limits in the event of a steam line rupture. The dose calculations for an assumed steam line rupture include the effects of a coincident 1.0 GPM primary to secondary tube leak in the steam generator of the affected steam line and a concurrent loss of offsite electrical power. These values are consistent with the assumptions used in the accident analyses.

Insert 1

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in HOT STANDBY within 6 hours and in HOT SHUTDOWN within the next 6 hours.
b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures ≤ 281°F within the next 6 hours.

Insert 2

Verify each pressurizer code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ±1%.

Insert 3

Surveillance Requirements are specified in the Inservice Testing Program. Pressurizer code safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code, which provides the activities and the frequency necessary to satisfy the Surveillance Requirements. No additional requirements are specified.

The pressurizer code safety valve as found setpoint is 2500 psia +3/-2.5% for OPERABILITY; however, the valves are reset to 2500 psia ±1% during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

Insert 4

Verify each main steam line code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ±1%.

Insert 5

Surveillance Requirement 4.7.1.1 verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The MSSV setpoints are 1000 psia +1/-3% (4 valves each header) and 1040 psia +1/-3% (4 valves each header) for OPERABILITY; however, the valves are reset to 1000 psia ±1% and 1040 psia ±1% respectively during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

The provisions of Specification 3.0.4 do not apply. This allows entry into and operation in MODE 3 prior to performing the Surveillance Requirement so that the MSSVs may be tested under hot conditions.

Attachment 4 to FPL Letter L-2000-001

ST. LUCIE UNIT 2 MARKED UP TECHNICAL SPECIFICATION PAGES

TS Page VI

TS Page 3/4 4-7

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TS Page B-3/4 4-2

TS Page 3/4 7-1

TS Page 3/4 7-3

TS Page B-3/4 7-1

INDEX

LIMITING CONDITION FOR OPERATION AND SURVEILLANCE REQUIREMENTS

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer code safety valve shall be OPERABLE with a lift setting of 2500 psia ± 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.

Deleted

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional Surveillance Requirements other than those required by the Inservice Testing Program.

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer code safety valves shall be OPERABLE with ~~a~~ as-left lift setting s ~~of 2500 psia ±1%.~~* ≥ 2460.3 psig and ≤ 2510.3 psig

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one pressurizer code safety valve inoperable, either restore the inoperable valve to OPERABLE status within 15 minutes or be in ~~at least~~ HOT STANDBY within 6 hours and in HOT SHUTDOWN within the ~~following~~ next 6 hours.

* Insert 1

SURVEILLANCE REQUIREMENTS

4.4.2.2 ~~No additional Surveillance Requirements other than those required by the Inservice Testing Program.~~

Insert 2

* The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

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 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 the Inservice Testing Program.~~

Insert 3

3/4.4.3 PRESSURIZER

An OPERABLE pressurizer provides pressure control for the Reactor Coolant System during operations with both forced reactor coolant flow and with natural circulation flow. The minimum water level in the pressurizer assures the pressurizer heaters, which are required to achieve and maintain pressure control, remain covered with water to prevent failure, which could occur if the heaters were energized uncovered. The maximum water level in the pressurizer ensures that this parameter is maintained within the envelope of operation assumed in the safety analysis. The maximum water level also ensures that the RCS is not a hydraulically solid system and that a steam bubble will be provided to accommodate pressure surges during operation. The steam bubble also protects the pressurizer code safety valves against water relief. The requirement to verify that on an Engineered Safety Features Actuation test signal concurrent with a loss of offsite power the pressurizer heaters are automatically shed from the emergency power sources is to ensure that the non-Class 1E heaters do not reduce the reliability of or overload the emergency power source. The requirement that a minimum number of pressurizer heaters be OPERABLE enhances the capability to control Reactor Coolant System pressure and establish and maintain natural circulation.

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line code safety valves shall be OPERABLE ~~with lift settings and orifice sizes~~ as ~~shown~~ specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With ~~both reactor coolant loops and associated steam generators in operation and with~~ one or more main steam line code safety valves inoperable, operation in MODES 1, 2 and 3 may proceed provided that, within 4 hours, either the inoperable valve is restored to OPERABLE status or the Power Level-High trip setpoint is reduced per Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in ~~COLD~~ HOT SHUTDOWN within the following ~~30~~ 12 hours.

b. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.7.1.1 ~~No additional Surveillance Requirements other than those required by the Inservice Testing Program.~~

Insert 4

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	Header A	Header B	AS-LEFT LIFT SETTING ~~(± 1%)~~	~~ORIFICE SIZE~~
a.	8201	8205	~~1000 psia~~ ≥ 975.3 psig and ≤ 995.3 psig	~~16 in.^{2}~~
b.	8202	8206	~~1000 psia~~ "	~~16 in.^{2}~~
c.	8203	8207	~~1000 psia~~ "	~~16 in.^{2}~~
d.	8204	8208	~~1000 psia~~ "	~~16 in.^{2}~~
e.	8209	8213	~~1040 psia~~ ≥ 1014.9 psig and ≤ 1035.7 psig	~~16 in.^{2}~~
f.	8210	8214	~~1040 psia~~ "	~~16 in.^{2}~~
g.	8211	8215	~~1040 psia~~ "	~~16 in.^{2}~~
h.	8212	8216	~~1040 psia~~ "	~~16 in.^{2}~~

3/4.7 PLANT SYSTEMS

BASES

3/4.7.1 TURBINE CYCLE

3/4.7.1.1 SAFETY VALVES

The OPERABILITY of the main steam line code safety valves ensures that the secondary system pressure will be limited to within 110% (1100 psia) of its design pressure of 1000 psia during the most severe anticipated system operational transient. The maximum relieving capacity is associated with a turbine trip from 100% RATED THERMAL POWER coincident with an assumed loss of condenser heat sink (i.e., no steam bypass to the condenser).

The specified valve lift settings and relieving capacities are in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1971 Edition, and ASME Code for Pumps and Valves, Class II. The total relieving capacity for all valves on all of the steam lines is 12.49×10^6 lbs/hr which is 103.8% of the total secondary steam flow of 12.03×10^6 lbs/hr at 100% RATED THERMAL POWER. A minimum of two OPERABLE safety valves per steam generator ensures that sufficient relieving capacity is available for removing decay heat.

STARTUP and/or POWER OPERATION is allowable with safety valves inoperable within the limitations of the ACTION requirements on the basis of the reduction in secondary system steam flow and THERMAL POWER required by the reduced reactor trip settings of the Power Level-High channels. The reactor trip setpoint reductions are derived on the following bases:

For two loop operation:

$$SP = \left[\frac{(X) - (Y)(V)}{X} \times (107.0) \right] - 0.9$$

where:

SP = reduced reactor trip setpoint in percent of RATED THERMAL POWER

V = maximum number of inoperable safety valves per steam line

107.0 = Power Level-High Trip Setpoint for two loop operation

0.9 = Equipment processing uncertainty

X = Total relieving capacity of all safety valves per steam line in lbs/hour (6.247×10^6 lbs/hr)

Insert 5

Y = Maximum relieving capacity of any one safety valve in lbs/hour (7.74×10^5 lbs/hr)

Insert 1

b. With two or more pressurizer code safety valves inoperable, be in HOT STANDBY within 6 hours and in HOT SHUTDOWN with all RCS cold leg temperatures ≤ 230°F within the next 6 hours.

Insert 2

Verify each pressurizer code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ±1%.

Insert 3

Surveillance Requirements are specified in the Inservice Testing Program. Pressurizer code safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code, which provides the activities and the frequency necessary to satisfy the Surveillance Requirements. No additional requirements are specified.

The pressurizer code safety valve as found setpoint is 2500 psia ±2% for OPERABILITY; however, the valves are reset to 2500 psia ±1% during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

Insert 4

Verify each main steam line code safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ±1%.

Insert 5

Surveillance Requirement 4.7.1.1 verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The MSSV setpoints are 1000 psia +1/-3% (4 valves each header) and 1040 psia +1/-3% (4 valves each header) for OPERABILITY; however, the valves are reset to 1000 psia ±1% and 1040 psia ±1% respectively during the Surveillance to allow for drift. The LCO is expressed in units of psig for consistency with implementing procedures.

The provisions of Specification 3.0.4 do not apply. This allows entry into and operation in MODE 3 prior to performing the Surveillance Requirement so that the MSSVs may be tested under hot conditions

11/17/99

See Proposed Change To Tech Specs.

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Subject:
St Lucie Unit 1 and Unit 2 Proposed License Amendments EDG Risk Informed AOT Exte nsion Evaluatin of proposed Technical Specifications changes enclosed.

Body:
pdr adock 05000335 p

Docket: 05000335, Notes: N/A

Docket: 05000389, Notes: N/A

AA2

FPL

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

November 17, 1999

L-99-228
10 CFR 50.90

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: St. Lucie Unit 1 and Unit 2
Docket Nos. 50-335 and 50-389
Proposed License Amendments
EDG Risk Informed AOT Extension

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 and NPF-16 for St. Lucie Unit 1 and Unit 2, respectively, by incorporating the attached Technical Specifications (TS) revisions. The amendments will revise the current 72-hour action completion time/allowed outage time (AOT) specified in TS 3.8.1.1, Action "b," to allow 14 days to restore an inoperable emergency diesel generator set to operable status. The proposed AOT is based on a cooperative study performed by participating members of the Combustion Engineering Owners Group (CEOG) in conjunction with supplemental information provided herein. The study included an integrated review and assessment of plant operations, deterministic design basis factors, and an evaluation of overall plant risk using probabilistic safety assessment (PSA) techniques. It is requested that the proposed amendment, if approved, be issued by August 1, 2000.

Attachment 1 is an evaluation of the proposed TS changes. Attachment 2 is the "Determination of No Significant Hazards Consideration." Attachments 3 and 4 contain copies of the appropriate TS pages marked-up to show the proposed changes. The St. Lucie Facility Review Group and the Florida Power & Light Company Nuclear Review Board have reviewed the proposed amendments, and a copy of this submittal is being forwarded to the State Designee for the State of Florida in accordance with 10 CFR 50.91 (b)(1).

Please contact us if there are any questions about this submittal.

Very truly yours,

JAStall

J. A. Stall
Vice President
St. Lucie Plant

JAS/RLD

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. W.A. Passetti, Florida Department of Health and Rehabilitative Services

A001

9912270248

PDR ADOCK 05000335

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

J. A. Stall being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

JASK

J. A. Stall

STATE OF FLORIDA

COUNTY OF St. Lucie

Sworn to and subscribed before me

this 17th day of November, 1999

by J. A. Stall, who is personally known to me.

Leslie J. Whitwell

Signature of Notary Public-State of Florida

Leslie J. Whitwell
MY COMMISSION # CC646183 EXPIRES
May 12, 2001
BONDED THRU TROY FAIN INSURANCE, INC.

Name of Notary Public (Print, Type, or Stamp)

11/17/99

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Subject:
Application for Amend to Licenses DPR-67 and NPF-16, Revising Tech Specs to Require Laboratory Testing of Activated Charcoal Samples for Applicable Engineered Safety Fe ature (ESF) Systems Using the ASTM D3803-1989 Protocol.

Body:
PDR ADOCK 05000335 P

Docket: 05000335, Notes: N/A

Docket: 05000389, Notes: N/A

AA2

FPL

Florida Power & Light Company, 6351 S. Ocean Drive, Jensen Beach, FL 34957

November 17, 1999

L-99-241
10 CFR 50.90
10 CFR 50.4

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Proposed License Amendments
GL 99-02 Charcoal Adsorber Test Protocol

Pursuant to 10 CFR 50.90, Florida Power & Light Company (FPL) requests to amend Facility Operating Licenses DPR-67 and NPF-16 for St. Lucie Unit 1 and Unit 2, respectively, by incorporating the attached Technical Specifications (TS) revisions. The amendments will revise the St. Lucie Unit 1 and 2 TS to require laboratory testing of activated charcoal samples for applicable engineered safety feature (ESF) ventilation systems using the ASTM D3803-1989 protocol.

The affected TS are Units 1 and 2 shield building ventilation system, TS 4.6.6.1; Unit 1 ECCS area ventilation system, TS 4.7.8.1; Unit 1 control room emergency ventilation system, TS 4.7.7.1; Unit 2 control room emergency air cleanup system, TS 4.7.7; and Unit 1 fuel pool ventilation system - fuel storage, TS 4.9.12. It is requested that the proposed amendments, if approved, be issued by February 15, 2000 to support testing prior to St. Lucie Unit 2 refueling outage (SL2-12) currently scheduled to begin in April, 2000.

Attachment 1 is an evaluation of the proposed TS changes. Attachment 2 is the "Determination of No Significant Hazards Consideration." Attachments 3 and 4 contain copies of the appropriate TS pages marked-up to show the proposed changes. The St. Lucie Facility Review Group and the FPL Company Nuclear Review Board have reviewed the proposed amendments. A copy of this submittal is being forwarded to the State Designee for the State of Florida in accordance with 10 CFR 50.91 (b)(1).

This letter implements the TS portion of GL 99-02 requested action 2 as committed in FPL letter L-99-232. FPL committed to submit the GL 99-02 requested license amendments by November 30, 1999. With this submittal, FPL satisfies the enforcement discretion guidance of GL 99-02. In the event that the staff does not approve the proposed license amendments by February 15, 2000, FPL is hereby requesting the NRC grant enforcement discretion in

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accordance with the guidance provided in GL 99-02 and the technical basis described in this submittal

In summary, this submittal states clearly FPL's intent to test to ASTM D3803-1989 if the TS are approved, otherwise enforcement discretion is expected from the NRC to allow St. Lucie to test to the ASTM D3803-1989 standard for the upcoming charcoal adsorber testing in the spring 2000.

Please contact us if there are any questions about this submittal.

Very truly yours,

J. A. Stall
Vice President
St. Lucie Plant

JAS/GRM

Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant
Mr. W. A. Passetti, Florida Department of Health and Rehabilitative Services

STATE OF FLORIDA)
) ss.
COUNTY OF ST. LUCIE)

J. A. Stall being first duly sworn, deposes and says:

That he is Vice President, St. Lucie Plant, for the Nuclear Division of Florida Power & Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

J. A. Stall

STATE OF FLORIDA

COUNTY OF St. Lucie

Sworn to and subscribed before me

this 17 day of November, 1999

by J. A. Stall, who is personally known to me.

Leslie J. Whitwell
Signature of Notary Public-State of Florida

Leslie J. Whitwell
MY COMMISSION # CC646183 EXPIRES
May 12, 2001
BONDED THRU TROY FAIN INSURANCE, INC.

Name of Notary Public (Print, Type, or Stamp)

ATTACHMENT 1

EVALUATION OF PROPOSED TS CHANGES

EVALUATION OF PROPOSED TS CHANGES

1.0 Introduction

The proposed amendments to Facility Operating Licenses DPR-67 for St. Lucie Unit 1 (PSL1) and NPF-16 for St. Lucie Unit 2 (PSL2) will revise the Unit 1 and 2 license requirements for laboratory testing the ability of activated charcoal samples to remove radioactive methyl iodide (organic iodine form) when the sample is tested in accordance with ASTM D3803-1989 requirements for the following ventilation systems:

Table 1 – Ventilation Systems with TS Requirements for Activated Charcoal Adsorber Testing

Unit 1	Unit 2
Shield Building Ventilation System (SBVS), TS 4.6.6.1	Shield Building Ventilation System(SBVS), TS 4.6.6.1
Control Room Emergency Ventilation System (CREVS), TS 4.7.7.1	Control Room Emergency Air Cleanup System (CREACS), TS 4.7.7
ECCS Area Ventilation System, TS 4.7.8.1	N/A For Unit 2, no credit for the ECCS Area filtration and cleanup systems in the accident analysis.
Fuel Handling Building Ventilation System (FHBVS), TS 4.9.12	N/A For Unit 2, the SBVS performs the filtration function

On June 3, 1999, the NRC issued Generic Letter (GL) 99-02, *Laboratory Testing of Nuclear-Grade Activated Charcoal.* This GL was issued to alert licensees of the NRC determination that testing nuclear-grade activated charcoal to standards other than American Society of Testing and Materials (ASTM) D3803-1989, *Standard Test Method for Nuclear-Grade Activated Carbon*, does not provide assurance for complying with the current licensing basis as it relates to the dose limits of GDC 19 of 10 CFR 50, Appendix A, or 10 CFR 100 Subpart A.

The GL also requested that all licensees determine whether their Technical Specifications (TS) reference ASTM D3803-1989 for charcoal laboratory testing. Licensees, whose TS do not reference this standard, should either amend the TS to reference the standard or propose an alternative test protocol. The St. Lucie Unit 1 and Unit 2 TS do not currently reference this standard for the test protocol.

The St. Lucie units have been testing charcoal samples in accordance with the TS referencing the following standards: ANSI N510-1975 for Unit 1, and ANSI N510-1980 for Unit 2. The St. Lucie Unit 1 and Unit 2 TS do not reference R.G. 1.52 for testing. GL 99-02 action 2 requested Licensees to submit TS amendment requests within 180 days of June 3, 1999. The amendment request should adopt the ASTM D3803-1989 test protocol and should contain the test temperature, relative humidity (RH), and penetration at which the proposed TS will require the test to be performed and the basis for these values.

2.0 Background

Safety related air-cleaning units used in plant ventilation systems reduce the potential onsite and offsite consequences of radiological accidents by adsorbing radioiodine. Analyses of design bases accidents assume particular safety related charcoal adsorption efficiencies when calculating offsite and control room operator doses. Licensees then test the charcoal to determine whether the adsorber efficiency is greater than that assumed in the design basis accident analysis. To ensure that the charcoal adsorbers will perform in a manner that is consistent with the licensing basis, most licensees have requirements in their TS to periodically test samples of activated charcoal used in these ventilation systems.

The industry and NRC position on the appropriate laboratory tests for nuclear-grade charcoal has evolved over the years since the issuance of Regulatory Guide (RG) 1.52, *Design, Testing, and Maintenance Criteria for Post-Accident Engineered Safety Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants*, which is referenced in many plant TS. It was initially assumed that high-temperature/high-relative-humidity conditions were the most severe. Later, it became clear that the most conservative test is at low temperature and high humidity. The use of outdated test protocols or inappropriate test conditions can lead to an overestimate of the charcoal ability to adsorb radioiodine following an accident.

GL 99-02 classified the nuclear plants in four categories based on NRC sponsored surveys:

1. Plants in compliance with their TS that test in accordance with ASTM D3803-1989,
2. Plants in compliance with their TS that test in accordance with test protocols other than ASTM D3803-1989,
3. Plants not in compliance with their TS that test in accordance with ASTM D3803-1989, and
4. Plants not in compliance with their TS that test in accordance with test protocols other than ASTM D3803-1989.

The St. Lucie units have been testing charcoal samples in accordance with the TS referencing the following standards: ANSI N510-1975 for Unit 1, and ANSI N510-1980 for Unit 2. Therefore, St. Lucie fits the GL 99-02 Category 2 designation. The St. Lucie Unit 1 and Unit 2 TS do not reference Regulatory Guide (RG) 1.52 for testing. However, both units were designed to RG 1.52 requirements as stated in the Updated Final Safety Analysis Report (UFSAR).

The shield building ventilation systems for both units have been designed to provide fission product removal capacity during design basis accident conditions. The SBVS consists of two full capacity redundant fan and filter subsystems, which share a common shield building duct intake and a common plant vent. Each filter subsystem consists of demisters, electric-heating coils, HEPA filters, and charcoal adsorbers enclosed in a common casing.

The control room emergency ventilation systems for both units have been designed to limit airborne radioactivity in the control room following a LOCA by recirculating control room air through charcoal adsorbers. This is required so that airborne radiological doses experienced by control room personnel following a design basis accident (DBA) do not exceed limits imposed by General Design Criterion 19. The control room ventilation system in both units are similar and they consist of split system air conditioners (i.e., indoor and outdoor section), a ducted air intake and air distribution system, and a filter train with HEPA filters and charcoal adsorbers with two redundant booster centrifugal fans.

The Unit 1 emergency core cooling system (ECCS) area ventilation system is designed to provide post-LOCA filtration and adsorption of fission products in the exhaust air. Also, it limits the post accident radiological doses below the guidelines of 10 CFR 100 from areas of the reactor auxiliary building which contain the following equipment:

a) Containment isolation valves,
b) Low pressure safety injection pumps,
c) High-pressure safety injection pumps,
d) Containment spray pumps,
e) Shutdown heat exchangers, and
f) Piping which contains recirculating containment sump water following a LOCA.

This ventilation system consists of two redundant centrifugal exhaust fans, HEPA and charcoal adsorbers, and associated ductwork, dampers and controls. The exhausted air is vented to the outside atmosphere.

The Fuel handling building (FHB) ventilation systems for both units serve the spent fuel pool areas. The Unit 1 FHB ventilation system is designed to reduce plant personnel doses by preventing the accumulation of airborne radioactivity in the fuel handling building due to diffusion of fission products from the spent fuel pool. This system is also designed to ventilate the spent fuel cooling equipment contained within the fuel handling building. This ventilation system consists of two separate supply systems and two separate exhaust systems. Each supply system consists of a hooded wall intake, an air handling unit with filters, a fan section, and a duct distribution system. One system supplies air to the fuel pool area and the other system supplies air to the lower areas. The fuel pool area air is exhausted through air inlets around the periphery of the fuel pool. Air is discharged by two 100 percent capacity centrifugal fans to the atmosphere through a prefilter, HEPA filter bank, charcoal absorbers and out the FHB vent stack. Air exhaust from the lower areas is passed through a prefilter and HEPA filter bank before being discharged by a centrifugal fan to the atmosphere through the FHB vent stack.

The portion of Unit 2 FHB ventilation system for the spent fuel pool ventilation is interconnected with the SBVS. In the event of a high radiation signal, the fuel pool area is exhausted to the plant vent via the SBVS filters.

3.0 Proposed TS Changes: Description and Basis

A. St. Lucie Unit 1 Shield Building Ventilation System:

1. Surveillance Requirement 4.6.6.1.b.3 – Change *ANSI N510-1975 (130^{0}C, 95% R.G.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

2. Surveillance Requirement 4.6.6.1.c.1 – Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

3. Surveillance Requirement 4.6.6.1.c.2 – Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

B. St. Lucie Unit 1 Control Room Emergency Ventilation System:

1. Surveillance Requirement 4.7.7.1.c.3 - Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

2. Surveillance Requirement 4.7.7.1.d.1 - Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

3. Surveillance Requirement 4.7.7.1.d.2 - Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

C. St. Lucie Unit 1 ECCS Area Ventilation System:

1. Surveillance Requirement 4.7.8.1.b.3 - Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

2. Surveillance Requirement 4.7.8.1.c.1 - Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

3. Surveillance Requirement 4.7.8.1.c.2 - Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

D. St. Lucie Unit 1 Fuel Pool Ventilation System – Spent Fuel:

1. Surveillance Requirement 4.9.12.b.3 - Change *ANSI N510-1975 (130^{0}C, 95% R.H.)* to *ASTM D3803-1989 (30^{0}C, 95% RH)*.

2. Surveillance Requirement 4.9.12.b.3 – Change removal efficiency of $\geq$ *70%* to $\geq$ *85%* and change ***elemental iodide*** to ***methyl iodide***.

E. St. Lucie Unit 2 Shield Building Ventilation System:

1. Surveillance Requirement 4.6.6.1.c - Change *ANSI N510-1980 (130°C, 95% R.H.)* to *ASTM D3803-1989 (30°C, 95% RH).*

2. Surveillance Requirement 4.6.6.1.c – Delete the following phrase *and ≥ 99% for radioactive elemental iodine.*

F. St. Lucie Unit 2 Control Room Emergency Air Cleanup system (CREACS):

1 Surveillance Requirement 4.7.7.d - Change *ANSI N510-1980 (130°C, 95% R.H.)* to *ASTM D3803-1989 (30°C, 95% RH).*

The current charcoal adsorber testing protocol used for Unit 1 per ANSI N510-1975 is performed in accordance with Table 4 of RTD M 16-1[1]. According to this table, charcoal testing for radioactive methyl iodide at 130°C and 95% relative humidity is performed as follows:

> *The test sample is pre-equilibrated with air at the above conditions until the temperature differential across the test bed is less than 1°C. With continuing air flow, steam is injected adjusting it to maintain partial pressure of air at 20 +/- 2 psia and steam at 95 +/- 1% of the saturated water pressure at the measured test temperature. The challenge gas (methyl iodide) is injected for 1.5 hrs. After iodine injection is stopped, steam-air injection is continued for another 1.5 hrs. Then, the test bed is allowed to cool, removed, and count tested to calculate efficiency.*

The current charcoal adsorber testing protocol used for Unit 2 per ANSI N510-1980 is performed in accordance with ASTM D3803 (1979). According to this standard, charcoal testing for radioactive methyl iodide at 130°C and 95% relative humidity is performed as follows:

> *The conditions are maintained for an equilibrium period of at least two hours, or until the temperature differential across the bed is less than 2°C. These conditions are continued during the challenging period when a specific amount of methyl iodide is added to the test gas flow for one hour. After this period, the flow conditions are maintained for a 240-minute elution period without the addition of the methyl iodide. A substantial amount of condensed water is collected during the test, and any radioactivity contained in it is assumed to have penetrated the carbon bed. For tests of used charcoal, the carbon sample*

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[1] U.S. AEC, Division of Reactor Research and Development, RDT Standard M16-1T, "Gas-Phase Adsorbents for Trapping Radioactive Iodine and iodine Compounds", Oct. 1973.

> *is not exposed to humid airflow for a pre-equilibration period. Instead, the bed is brought to the test temperature without airflow, followed by the challenging period.*

According to GL 99-02, studies by EG&G reported that this standard (ASTM D3803-1979) had unacceptable test parameter tolerances and instrument calibration requirements, and that the standard was non-conservative in not requiring humidity pre-equilibration of used charcoals.

The proposed charcoal adsorber-testing protocol (ASTM D3803-1989), which will be incorporated for both units per the proposed amendments, will be performed as follows:

> *The test sample is exposed to air at 30°C and 95% RH (1 atmosphere) for a pre-equilibrium period of 16 hours, followed by airflow at equilibrium for 2-hours. During the challenge period, methyl iodide is injected at a fixed mass concentration for 1-hour. This is followed by injection of humid air only for another hour (elution period). Throughout the entire test the effluent from the sample bed passes through two backup beds containing carbon having a known high efficiency for methyl iodide. The beds trap all methyl iodide that passes the test bed and provide a differential indication of their efficiency. At the end of the elution period, the I-131 gamma activity is measured and the percent of adsorbate penetrating the test bed is determined.*

The results of the ASTM D3803-1989 test protocol provide a conservative estimate of the performance of nuclear-grade activated charcoals used in all nuclear power plant ventilation systems for the removal of iodine. Also, according to this standard, the 30°C, 95% RH methyl iodide test is the most reliable test protocol to establish the methyl iodide removal efficiency of any adsorbent. The NRC has agreed with these conclusions as demonstrated by the fact that the ASTM D3803-1989 protocol is accepted and endorsed by the NRC in GL 99-02. Additionally, the Committee on Nuclear Air and Gas Treatment (CONAGT) and NRC-INEL discussions have concluded that the humidity pre-equilibration at 30°C for used charcoals results in a more conservative test than the non pre-equilibration required by previous versions of ASTM D3803. It is then concluded that the new proposed testing protocol per ASTM D3803-1989 at 30°C, 95% RH with different pre-equilibration, testing, and elution techniques which is now endorsed by the NRC and results in more conservative results than other test methods, is an acceptable methodology for the St. Lucie safety related ESF ventilation system charcoal testing.

The UFSAR discusses the need for charcoal testing of the safety related ventilation systems as defined in Table 1 of this attachment, and with the conservative methyl iodide penetration requirements stated in the Technical Specifications. However, testing of the Unit 2 ECCS ventilation system charcoal beds is not required from a licensing point of view as concluded in the NRC Initial Operating License Safety Evaluation Report (NUREG-0843). Also, the Unit 2 fuel pool ventilation system is interconnected to the SBVS, and during accident conditions, airflow from the fuel handling building is routed through the SBVS. For these reasons, the

Unit 2 ECCS and FHB ventilation systems are not required to be tested for methyl iodide penetration.

Based on the above information, the assumed accident analysis penetration values for each ESF ventilation system along with FPL proposed penetration values for methyl iodide testing, based on a minimum safety factor value of 2 is shown below:

Table 2 Allowable Values

ESF Ventilation System	Assumed Accident Analysis Penetration	GL 99-02 Maximum Allowable Penetration with SF = 2	Proposed Maximum Allowable Penetration	Proposed TS Minimum Allowable Efficiency
Unit 1 SBVS	30%	15%	10%	90%
Unit 1 ECCS Area Ventilation System	30%	15%	10%	90%
Unit 1 CREVS	30%	15%	10%	90%
Unit 1 FHBVS	30%	15%	15%	85%
Unit 2 SBVS	50%	25%	10%	90%
Unit 2 CREACS	1%	0.5%	0.175%	99.825%

From above, it is clear that the penetration values proposed by this proposed amendment are the same as or more conservative than the values correlated to a safety factor of 2 which is the minimum value allowed by GL 99-02. (Safety factor is defined as the ratio between the penetration value assumed in the accident analysis to the allowable testing value). The proposed penetration values are equivalent to charcoal adsorber efficiencies provided above and in the TS markups (Attachments 3 and 4). The maximum allowable penetration values in the proposed TS will be kept the same as the values currently provided in the TS for all the systems except the Unit 1 FHBVS. This will provide additional conservatism. For the Unit 1 FHBVS the proposed maximum allowable penetration value basis is discussed below. The decision to maintain the same penetration values was based on maintaining the same level of conservatism currently established while providing operating flexibility. According to the charcoal testing vendor, for the most limiting case shown above (Unit 2 CREACS) which has a higher safety factor than 2, this translates into a charcoal adsorber availability for several years without being changed out. This is based on testing of charcoal adsorbers from other facilities similar to St. Lucie.

Since a methyl iodide testing criteria is currently not provided for the fuel handling building ventilation for Unit 1 (existing TS only test for elemental iodine), a penetration value for methyl iodide testing is being proposed for this specification. The proposed penetration value is based on the 30% penetration assumed in the accident analysis and applying a safety factor of 2, the minimum value allowed by GL 99-02. Therefore, a penetration testing criteria of 15% will be incorporated into the Technical Specifications for this system, equivalent to charcoal adsorber efficiency of 85%.

With regards to the proposed testing relative humidity (95%), this value is also more conservative than the value assumed in the accident analysis. Normally, a relative humidity (RH) of 70% is maintained at all times (including post-accident) by the SBVS in both units. This is achieved by redundant electrical heaters provided for each train of SBVS. For the other ESF ventilation systems located outside containment, the UFSARs assume that based on system configurations, it is not expected that the relative humidity for charcoal adsorbers will be higher than 70% even without the use of heaters. The proposed RH value (95%) is much larger than 70% and it is more conservative because increasing the RH of the test generally decreases the efficiency of methyl iodide removal by activated carbon.

The testing temperature for the proposed TS will be 30°C instead of the current value of 130°C, to be consistent with the recommendations provided in GL 99-02.

In addition to the proposed Technical Specification changes from above, the requirement to perform elemental iodine testing will be removed from applicable TS to be consistent with GL 99-02 guidance that requires only methyl iodide testing for charcoal adsorber efficiency.

There is no safety significance associated with implementing a more conservative charcoal testing protocol than previous tests. The new testing methodology will eliminate the potential for having overestimation of the charcoal bed efficiency. Therefore, testing at lower temperatures provides for a more conservative approach for the charcoal to react to the plant conditions existing during design basis accidents. There are no modifications to safety-related equipment or adverse effects imposed by the new testing methodology. Therefore, all safety related structures, systems, and components remain unaffected and still capable of performing their design basis functions. The St. Lucie plant complies with 10 CFR 50 Appendix A, General Design Criterion 19 and 10 CFR 100, Subpart A.

4.0 Environmental Consideration

The proposed license amendments change requirements with respect to surveillance requirements. The amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite, and no significant increase in individual or cumulative occupational radiation exposure. FPL has concluded that the proposed amendments involve no significant hazards consideration and meet the criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). FPL has determined pursuant to 10 CFR 51.22(b), that an environmental impact statement or environmental assessment need not be prepared in connection with issuance of the amendments

5.0 Conclusion

Based on review of the licensing bases documentation and evaluation presented above, the new testing protocol and testing conditions proposed by this evaluation provide a more conservative approach to laboratory testing of charcoal adsorbers for safety related ventilation systems. The requirements of General Design Criterion 19 and 10 CFR 100 Subpart A are maintained. The proposed new testing methodology will ensure the design basis of each plant (relative to methyl iodide removal) is maintained and that safety analysis assumptions remain valid.

ATTACHMENT 2

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

Description of amendment request:

The amendments will revise the St. Lucie Unit 1 and 2 TS to require laboratory testing of activated charcoal samples for applicable engineered safety feature (ESF) ventilation systems using the ASTM D3803-1989 protocol. The affected TS are Units 1 and 2 shield building ventilation system, TS 4.6.6.1; Unit 1 ECCS area ventilation system, TS 4.7.8.1; Unit 1 control room emergency ventilation system, TS 4.7.7.1; Unit 2 control room emergency air cleanup system, TS 4.7.7; and Unit 1 fuel pool ventilation system - fuel storage, TS 4.9.12.

Pursuant to 10 CFR 50.92, a determination may be made that a proposed license amendment involves no significant hazards consideration if operation of the facility in accordance with the proposed amendment would not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. Each standard is discussed as follows:

(1) Operation of the facility in accordance with the proposed amendment would not involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed amendment does not involve a significant increase in the probability or consequences of any accident previously evaluated. The new charcoal testing protocol is performed offsite on samples extracted from the safety-related ventilation systems. Therefore, there is no impact on any accident initiator and therefore, no changes on the probability. The proposed testing protocol is more conservative than previous tests; therefore, the efficiency of charcoal for the affected safety-related systems would not be overestimated. With the new testing protocol, more conservative testing results are expected since the temperature at which testing is performed is lower and the charcoal retention capability is more consistent with actual accident conditions. The proposed change thus ensures that the charcoal in service will comply with the penetration requirements to meet the design basis accident conditions.

Therefore, operation of the facility in accordance with the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated. The proposed new charcoal testing protocol only affects surveillance testing requirements for ventilation systems. The functions of these

systems remain unchanged and unaffected. No new system interactions have been introduced by the proposed amendment, which would create a new or different type of accident than previously analyzed. No physical changes are being made to any structure, system or component. The operation of the facility has not been altered by the proposed amendment. The systems involved are not considered to initiate any accidents as previously evaluated.

The proposed amendment will not change the physical plant or the modes of operation defined in the facility license. The changes do not involve the addition of new equipment or the modification of existing equipment, nor do they alter the design of St. Lucie plant systems. Therefore, operation of the facility in accordance with the proposed amendment would not create the possibility of a new or different kind of accident from any accident previously evaluated.

(3) Operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

The proposed amendment does not involve a reduction in the margin of safety. The margin of safety of the Technical Specifications, its bases, the Final Safety Analysis Report, the Safety Evaluation Report or in any other design document has not been affected by the proposed amendment. The change provided in this proposed amendment is related to introducing an improved testing protocol for the activated charcoal in safety related ventilation systems. The change consists of testing the charcoal with a new testing protocol and with lower test temperatures to resemble more closely accident conditions and to eliminate potential overestimation of charcoal efficiency.

Therefore, operation of the facility in accordance with the proposed amendment would not involve a significant reduction in a margin of safety.

Based on the discussion presented above and on the supporting evaluation of proposed TS changes, FPL has concluded that these proposed license amendments involve no significant hazards consideration.

ATTACHMENT 3

ST. LUCIE UNIT 1 MARKED-UP TECHNICAL SPECIFICATION PAGES

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested accordance with ~~ANSI N510-1975 (130°C, 95% R.G.)~~. The carbon samples not obtained from test canisters shall be prepared by either:

ASTM D3803-1989 (30°C, 95% RH)

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 6000 cfm $\pm$ 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of system operation by either:

1. Verifying that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the sample is tested in accordance with ~~ANSI N510-1975 (130°C, 95% R.H.)~~; or

2. Verifying that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of $\geq$ 90% for radioactive methyl iodide when the samples are tested in accordance with ~~ANSI N510-1975 (130°C, 95% R.H.)~~ and the samples are prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7.1 The control room emergency ventilation system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is ≤ 120°F.

b. At least once per 31 days by:

1. Initiating flow through the HEPA filter and charcoal adsorber train and verifying that each booster fan operates for at least 15 minutes.

2. Starting (unless already operating) each air conditioning unit and verifying that it operates for at least 8 hours.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housing, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ~~ANSI N510-1975 (130°C, 95% R.H.)~~ ASTM D 3803-1989 (30°C, 95% RH). The carbon samples not obtained from test canisters shall be prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

ASTM D 3803-1989 (30°C, 95% RH)

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 2000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

ASTM D3803-1989 (30°C, 95% RH)

d. After every 720 hours of system operation by either:

1. Verifying that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ~~ANSI N510-1975 (130°C, 95% R.H.)~~; or

2. Verifying that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of ≥ 90% for radioactive methyl iodide when the samples are tested in accordance with ~~ANSI N510-1975 (130°C, 95% R.H.)~~ and the samples are prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

Subsequent to reinstalling the adsorber tray used for obtaining the carbon sample, the system shall be demonstrated OPERABLE by also:

a) Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%, and

b) Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 2000 cfm ± 10%.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ~~ANSI-N510-1975 (130°C, 95% R.H.)~~. The carbon samples not obtained from test canisters shall be prepared by either:

ASTM D3803-1989 (30°C, 95% RH)

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 30,000 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

c. After every 720 hours of system operation by either:

1. Verifying that a laboratory analysis of a carbon sample obtained from a test canister demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide when the sample is tested in accordance with ~~ANSI-N510-1975 (130°C, 95% R.H.)~~; or

2. Verifying that a laboratory analysis of at least two carbon samples demonstrate a removal efficiency of ≥ 90% for radioactive methyl iodide when the samples are tested in accordance with ~~ANSI-N510-1975 (130°C, 95% R.H.)~~ and the samples are prepared by either:

a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

REFUELING OPERATIONS

SURVEILLANCE REQUIREMENTS (Continued)

1. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

2. Verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the ventilation system at a flow rate of 10,350 cfm ± 10%.

3. Verifying that a laboratory analysis of a carbon sample from either at least one test canister or at least two carbon samples removed from one of the charcoal adsorbers demonstrates a removal efficiency of ~~≥ 70%~~ ≥ 85% for radioactive ~~elemental~~ methyl iodide when the sample is tested in accordance with ~~ANSI N510-1975 (130°C, 95% R.H.)~~ ASTM D3803-1989 (30°C, 95% RH). The carbon samples not obtained from test canisters shall be prepared by either:

   a) Emptying one entire bed from a removed adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed, or

   b) Emptying a longitudinal sample from an adsorber tray, mixing the adsorbent thoroughly, and obtaining samples at least two inches in diameter and with a length equal to the thickness of the bed.

4. Verifying a system flow rate of 10,350 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

ATTACHMENT 4

ST. LUCIE UNIT 2 MARKED-UP TECHNICAL SPECIFICATION PAGES

CONTAINMENT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

2. Performing airflow distribution to HEPA filters and charcoal adsorbers in accordance with ANSI N-510-1980. The distribution shall be ± 20% of the average flow per unit.

3. Verifying that the charcoal adsorbers remove ≥ 99% of a halogenated hydrocarbon refrigerant test gas when they are tested in place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

4. Verifying that the HEPA filter banks remove ≥ 99.825% of the DOP when they are tested in place in accordance with ANSI N-510-1980 while operating the system at a flow rate of 6000 cfm ± 10%.

5. Verifying a system flow rate of 6000 cfm ± 10% during system operation when tested in accordance with ANSI N-510-1980.

c. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a 2-inch laboratory sample from the installed sample canisters demonstrates a removal efficiency of ≥ 90% for radioactive methyl iodide ~~and ≥ 99% for radioactive elemental iodine~~ when tested in accordance with ~~ANSI N-510-1980 (130°C, 95% R.H.)~~. ASTM D3803-1989 (30°C, 95% RH)

d. At least once per 18 months by:

1. Verifying that the pressure drop across the demisters, electric heaters, HEPA filters, and charcoal adsorber banks is less than 8.5 inches Water Gauge (WG) while operating the system at a flow rate of 6000 cfm ± 10%.

2. Verifying that the system starts on a Unit 2 containment isolation signal and on a fuel pool high radiation signal.

3. Verifying that the filter cooling makeup and cross connection valves can be manually opened.

4. Verifying that each system produces a negative pressure of greater than or equal to 2.0 inches WG in the annulus within 99 seconds after a start signal.

5. Verifying that the main heaters dissipate 30 ± 3 kW and the auxiliary heaters dissipate 1.5 ± 0.25 kW when tested in accordance with ANSI N-510-1980.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

4.7.7 Each control room emergency air cleanup system shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the control room air temperature is less than or equal to 120°F.

b. At least once per 31 days by (1) initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 15 minutes and (2) starting, unless already operating each air conditioning unit and verifying that it operates for at least 8 hours.

c. At least once per 18 months or (1) after any structural maintenance on the HEPA filter or charcoal adsorber housings, or (2) following painting, fire or chemical release in any ventilation zone communicating with the system by:

1. Performing a visual examination of CREACS in accordance with ANSI N-510-1980.

2. Performing air flow distribution to HEPA filters and charcoal adsorbers in accordance with ANSI N-510-1980. The distribution shall be ± 20% of the average flow per unit.

3. Verifying that the charcoal adsorbers remove $\geq$ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in accordance with ANSI N-510-1980 while operating the system at 2000 cfm ± 10%.

4. Verifying that the HEPA filters remove $\geq$ 99.95% of DOP when they are tested in accordance with ANSI N-510-1980 while operating the system at 2000 cfm ± 10%.

5. Verifying a system flow rate of 2000 cfm ± 10%.

d. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a 4-inch laboratory sample from the installed sample canisters demonstrates a removal efficiency of $\geq$ 99.825% for methyl iodide when tested in accordance with ~~ANSI N-510-1980 at 130°C, 95% R.H.~~ ASTM D3803-1989 (30°C, 95% RH)

e. At least once per 18 months by:

1. Verifying that the pressure drop across the combined prefilters, HEPA filters and charcoal adsorber banks is less than 7.4 inches Water Gauge while operating the system at a flow rate of 2000 cfm ± 10%.

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(ASTM D3803-1989(30°C, 95% RH)) 6

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AA2

FPL

Florida Power & Light Company, P. O. Box 14000, Juno Beach, FL 33408-0420

NOV 0 8 1999
L-99-234
10 CFR Part 54

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Re: St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
Submittal Plan for License Renewal Application

The purpose of this letter is to inform the Nuclear Regulatory Commission and its staff of Florida Power and Light Company's (FPL) plans to submit a license renewal application for St. Lucie Units 1 and 2 by the end of June 2002. FPL is encouraged by the progress in the NRC's review of the Calvert Cliffs and Oconee license renewal applications and believes that license renewal can be accomplished for St. Lucie in an economical and timely manner.

FPL recognizes that the application for a renewed license for St. Lucie Unit 2 would be submitted more than 20 years prior to expiration of the existing license (2023). Accordingly, FPL will submit a request for exemption per 10 CFR Part 50.12 from the requirements of 10 CFR Part 54.17(c) by the end of 2000.

Should you have any questions concerning this letter, please contact Rajiv S. Kundalkar, Vice President, Nuclear Engineering at (561) 694-4848.

Sincerely yours,

Thomas F Plunkett

Thomas F. Plunkett
President
Nuclear Division

cc: Director, Office of Nuclear Reactor Regulation
Associate Director, Inspection and Programs
Division Director, Regulatory Improvement Programs
Chief, License Renewal and Standardization Branch
Project Manager, NRR, St. Lucie Plant
Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, St. Lucie Plant

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