

January 18, 1990

Docket Nos. 50-335
and 50-389

DISTRIBUTION
See attached sheet

Mr. J. H. Goldberg
Executive Vice President
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Goldberg:

The Commission has issued the enclosed Amendment Nos. 101 and 44 to Facility Operating License Nos. DPR-67 and NPF-16 for the St. Lucie Plant, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications in response to your application dated February 21, 1989.

These amendments change the iodine reporting requirements to conform with the recommendations of Generic Letter 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes."

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

Sincerely,

Original Signed By

Jan A. Norris, Senior Project Manager
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 101 to DPR-67
2. Amendment No. 44 to NPF-16
3. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDII-2	:PM:PDII-2	:D:PDII-2	:OGC	:PRPB	:	:
NAME	:DN:Ver	:JNorris	:HBerk	:JCN	:LCunningham	:	:
DATE	:1/18/90	:12/1/89	:1/5/90	:1/6/90	:1/1	:	:

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Document Name: ST. LUCIE AMENDMENT

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Mr. J. H. Goldberg
Florida Power & Light Company

St. Lucie Plant

cc:

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DATED: January 18, 1990

AMENDMENT NO. ¹⁰¹ TO FACILITY OPERATING LICENSE NO. DPR-67 - ST. LUCIE, UNIT 1
AMENDMENT NO. ⁴⁴ TO FACILITY OPERATING LICENSE NO. NPF-16 - ST. LUCIE, UNIT 2

Docket File

NRC & Local PDRs

PDII-2 Reading

S. Varga, 14/E/4

G. Lainas, 14/H/3

H. Berkow

D. Miller

J. Norris

OGC-WF

D. Hagan, 3302 MNBB

E. Jordan, 3302 MNBB

B. Grimes, 9/A/2

G. Hill (8), P1-137

Wanda Jones, P-130A

J. Calvo, 11/F/23

ACRS (10)

GPA/PA

OC/LFMB

PD Plant-specific file [Gray File]

M. Sinkule, R-II

Others as required

cc: Plant Service list



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

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-335

ST. LUCIE PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 101
License No. DPR-67

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated February 21, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

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

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 101, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION


for Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 18, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 101
TO FACILITY OPERATING LICENSE NO. DPR-67
DOCKET NO. 50-335

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

Remove Pages

3/4 4-17

3/4 4-18

B 3/4 4-5

6-16

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Insert Pages

3/4 4-17

3/4 4-18

B 3/4 4-5

6-16

6-16a

REACTOR COOLANT SYSTEM

SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

- a. $\leq 1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$, and
- b. $\leq 100/\bar{E} \mu\text{Ci/gram}$.

APPLICABILITY: MODES 1, 2, 3, 4 and 5.

ACTION:

MODES 1, 2 and 3*:

- a. With the specific activity of the primary coolant $>1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$ for more than 100 hours during one continuous time interval or exceeding the limit line shown on Figure 3.4-1, be in HOT STANDBY with $T_{\text{avg}} < 500^\circ\text{F}$ within 6 hours.
- b. With the specific activity of the primary coolant $>100/\bar{E} \mu\text{Ci/gram}$, be in HOT STANDBY with $T_{\text{avg}} < 500^\circ\text{F}$ within 6 hours.

MODES 1, 2, 3, 4 and 5:

With the specific activity of the primary coolant $>1.0 \mu\text{Ci/gram DOSE EQUIVALENT I-131}$ or $>100/\bar{E} \mu\text{Ci/gram}$, perform the sampling and analysis requirement of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

* With $T_{\text{avg}} \geq 500^\circ\text{F}$.

REACTOR COOLANT SYSTEM

DELETED

REACTOR COOLANT SYSTEM

BASES

time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 GPM and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the St. Lucie site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity > 1.0 $\mu\text{Ci}/\text{gram}$ DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

BASES

Reducing T_{avg} to $< 500^{\circ}F$ prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take correction action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2.1 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside surface and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location than at the outside surface location, the inside surface flaw may be more limiting. Consequently, for the heatup analysis, both the inside surface and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface.

Since neutron irradiation damage is also greater at the inside surface, the inside surface flaw location is the limiting location during cooldown. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

ADMINISTRATIVE CONTROLS

- (iv) Procedures for the recording and management of data,
- (v) Procedures defining corrective actions for all off-control point chemistry conditions, and
- (vi) A procedure identifying (a) the authority responsible for the interpretation of the data, and (b) the sequence and timing of administrative events required to initiate corrective action.

d. Backup Method for Determining Subcooling Margin

A program which will ensure the capability to accurately monitor the Reactor Coolant System subcooling margin. This program shall include the following:

- (i) Training of personnel, and
- (ii) Procedures for monitoring.

e. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

- (i) Training of personnel,
- (ii) Procedures for sampling and analysis, and
- (iii) Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the NRC.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an operating license, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant.

ADMINISTRATIVE CONTROLS

6.9.1.2 The startup report shall address each of the tests identified in the FSAR and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

ANNUAL REPORTS ^{1/}

6.9.1.4 Annual reports covering the activities of the unit as described below for the previous calendar year shall be submitted prior to March 1 of each year. The initial report shall be submitted prior to March 1 of the year following initial criticality.

6.9.1.5 Reports required on an annual basis shall include a tabulation on an annual basis of the number of station, utility and other personnel (including contractors) receiving exposures greater than 100 mrem/yr and their associated man-rem exposure according to work and job functions, ^{2/} e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

Annual reports shall also include the results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last

^{1/} A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station.

^{2/} This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ADMINISTRATIVE CONTROLS

isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC, no later than the 15th of each month following the calendar month covered by the report.



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

FLORIDA POWER & LIGHT COMPANY

ORLANDO UTILITIES COMMISSION OF

THE CITY OF ORLANDO, FLORIDA

AND

FLORIDA MUNICIPAL POWER AGENCY

DOCKET NO. 50-389

ST. LUCIE PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. NPF-16

1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by Florida Power & Light Company, et al. (the licensee), dated February 21, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

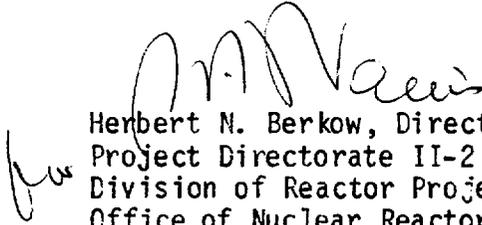
2. Accordingly, Facility Operating License No. NPF-16 is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and by amending paragraph 2.C.2 to read as follows:

2. Technical Specifications

- The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 44, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION


Herbert N. Berkow, Director
Project Directorate II-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 18, 1990

ATTACHMENT TO LICENSE AMENDMENT NO. 44

TO FACILITY OPERATING LICENSE NO. NPF-16

DOCKET NO. 50-389

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

Remove Pages

3/4 4-25

3/4 4-26

B 3/4 4-6

B 3/4 4-7

6-17

Insert Pages

3/4 4-25

3/4 4-26

B 3/4 4-6

B 3/4 4-7

6-17

REACTOR COOLANT SYSTEM

3/4.4.8 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

3.4.8 The specific activity of the primary coolant shall be limited to:

- a. Less than or equal to 1.0 microcurie/gram DOSE EQUIVALENT I-131, and
- b. Less than or equal to $100/\bar{E}$ microcuries/gram.

APPLICABILITY: MODES 1, 2, 3, 4 and 5

ACTION:

MODES 1, 2 and 3*:

- a. With the specific activity of the primary coolant greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or exceeding the limit line shown in Figure 3.4-1, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.
- b. With the specific activity of the primary coolant greater than $100/\bar{E}$ microcuries/gram, be in at least HOT STANDBY with T_{avg} less than 500°F within 6 hours.

MODES 1, 2, 3, 4 and 5.

With the specific activity of the primary coolant greater than 1 microcurie/gram DOSE EQUIVALENT I-131 or greater than $100/\bar{E}$ microcuries/gram, perform the sampling and analysis requirements of item 4 a) of Table 4.4-4 until the specific activity of the primary coolant is restored to within its limits.

SURVEILLANCE REQUIREMENTS

4.4.8 The specific activity of the primary coolant shall be determined to be within the limits by performance of the sampling and analysis program of Table 4.4-4.

*With T_{avg} greater than or equal to 500°F.

REACTOR COOLANT SYSTEM

DELETED

REACTOR COOLANT SYSTEM

BASES

LEAKAGE DETECTION SYSTEMS (Continued)

Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

3/4.4.6.2 OPERATIONAL LEAKAGE

Industry experience has shown that while a limited amount of leakage is expected from the RCS, the unidentified portion of this leakage can be reduced to a threshold value of less than 1 gpm. This threshold value is sufficiently low to ensure early detection of additional leakage.

The 10 gpm IDENTIFIED LEAKAGE limitation provides allowances for a limited amount of leakage from known sources whose presence will not interfere with the detection of UNIDENTIFIED LEAKAGE by the leakage detection systems.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowable limit.

The total steam generator tube leakage limit of 1 gpm for all steam generators ensures that the dosage contribution from the tube leakage will be limited to a small fraction of Part 100 limits in the event of either a steam generator tube rupture or steam line break. The 1 gpm limit is consistent with the assumptions used in the analysis of these accidents. The 0.5 gpm leakage limit per steam generator ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions.

PRESSURE BOUNDARY LEAKAGE of any magnitude is unacceptable since it may be indicative of an impending gross failure of the pressure boundary. Therefore, the presence of any PRESSURE BOUNDARY LEAKAGE requires the unit to be promptly placed in COLD SHUTDOWN.

3/4.4.7 CHEMISTRY

The limitations on Reactor Coolant System chemistry ensure that corrosion of the Reactor Coolant System is minimized and reduces the potential for Reactor Coolant System leakage or failure due to stress corrosion. Maintaining

REACTOR COOLANT SYSTEM

BASES

CHEMISTRY (Continued)

the chemistry within the Steady State Limits provides adequate corrosion protection to ensure the structural integrity of the Reactor Coolant System over the life of the plant. The associated effects of exceeding the oxygen, chloride and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the Reactor Coolant System. The time interval permitting continued operation within the restrictions of the Transient Limits provides time for taking corrective actions to restore the contaminant concentrations to within the Steady State Limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.4.8 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the resulting 2 hour doses at the site boundary will not exceed an appropriately small fraction of Part 100 limits following a steam generator tube rupture accident in conjunction with an assumed steady state primary-to-secondary steam generator leakage rate of 1.0 gpm and a concurrent loss of offsite electrical power. The values for the limits on specific activity represent limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters of the St. Lucie site, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 1.0 microcurie/gram DOSE EQUIVALENT I-131, but within the allowable limit shown on Figure 3.4-1, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER.

Reducing T_{avg} to less than 500°F prevents the release of activity should a steam generator tube rupture since the saturation pressure of the primary coolant is below the lift pressure of the atmospheric steam relief valves. The surveillance requirements provide adequate assurance that excessive specific activity levels in the primary coolant will be detected in sufficient time to take corrective action. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analyses following power changes may be permissible if justified by the data obtained.

REACTOR COOLANT SYSTEM

BASES

SPECIFIC ACTIVITY (Continued)

DELETED

3/4.4.9 PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the FSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients through the reactor vessel wall produce thermal stresses which are compressive at the reactor vessel inside surface and which are tensile at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the outside surface location. However, since neutron irradiation damage is larger at the inside surface location when compared to the outside surface, the inside surface flaw may be more limiting. Consequently, for the heatup analysis both the inside and outside surface flaw locations must be analyzed for the specific pressure and thermal loadings to determine which is more limiting.

During cooldown, the thermal gradients through the reactor vessel wall produce thermal stresses which are tensile at the reactor vessel inside surface and which are compressive at the reactor vessel outside surface. Since reactor vessel internal pressure always produces tensile stresses at both the inside and outside surface locations, the total applied stress is greatest at the inside surface location. Since the neutron irradiation damage is also greatest at the inside surface location the inside surface flaw is the limiting location. Consequently, only the inside surface flaw must be evaluated for the cooldown analysis.

The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 are composite curves which were prepared by determining the most conservative case, with either the inside or outside wall controlling, for any heatup rate of up to 50 degrees F per hour or cooldown rate of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of the predicted adjusted reference temperature at the end of the service period, and they include adjustments for possible errors in the pressure and temperature sensing instruments.

The reactor vessel materials have been tested to determine their initial RT_{NDT} ; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation will cause an increase in the RT_{NDT} . An adjusted reference temperature can be predicted using a) the initial RT_{NDT} , b) the fluence (E greater than 1 MeV), including appropriate adjustments for neutron attenuation and neutron energy spectrum variations through the wall thickness, c) the copper and nickel contents of the material, and d) the transition temperature shift from the curve shown in Figure B 3/4.4-1 as recommended by Regulatory Guide 1.99, Revision 2 (Draft), "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves Figures 3.4-2, 3.4-3 and 3.4-4 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period.

ADMINISTRATIVE CONTROLS

ANNUAL REPORTS (Continued)

greater than 100 mrems/yr and their associated man-rem exposure according to work and job functions,^{2/} e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling. The dose assignments to various duty functions may be estimated based on pocket dosimeter, TLD, or film badge measurements. Small exposures totalling less than 20% of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources should be assigned to specific major work functions.

- b. The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 3.4.8. The following information shall be included: (1) Reactor power history starting 48 hours prior to the first sample in which the limit was exceeded; (2) Results of the last isotopic analysis for radioiodine performed prior to exceeding the limit, results of analysis while the limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include date and time of sampling and the radioiodine concentrations; (3) Clean-up system flow history starting 48 hours prior to the first sample in which the limit was exceeded; (4) Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady-state level; and (5) The time duration when the specific activity of the primary coolant exceeded the radioiodine limit.

MONTHLY OPERATING REPORTS

6.9.1.6 Routine reports of operating statistics and shutdown experience, including documentation of all challenges to the PORVs or safety valves, shall be submitted on a monthly basis to the NRC, no later than the 15th of each month following the calendar month covered by the report.

^{2/} This tabulation supplements the requirements of §20.407 of 10 CFR Part 20.

ADMINISTRATIVE CONTROLS

SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT*

6.9.1.7 Routine Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year. The period of the first report shall begin with the date of initial criticality.

The Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof.

The Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of wind speed, wind direction, and atmospheric stability.** This same report shall include an assessment of the radiation doses due to the radioactive liquid and gaseous effluents released from the unit or station during the previous calendar year. This same report shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the SITE BOUNDARY (Figure 5.1-1) during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time and location, shall be included in these reports. The meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in the OFFSITE DOSE CALCULATION MANUAL (ODCM).

Every 2 years using the previous 6 months release history for isotopes and historical meteorological data determine the controlling age group for both liquid and gaseous pathways. If changed from current submit change to ODCM to reflect new tables for these groups and use the new groups in subsequent dose calculations.

The Radioactive Effluent Release Report to be submitted 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed MEMBER OF THE PUBLIC from reactor releases for the previous

* A single submittal may be made for a multiple unit station. The submittal should combine those sections that are common to all units at the station; however, for units with separate radwaste systems, the submittal shall specify the releases of radioactive material from each unit.

** In lieu of submission with the Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 101 AND 44

TO FACILITY OPERATING LICENSE NOS. DPR-67 AND NPF-16

FLORIDA POWER & LIGHT COMPANY, ET AL.

ST. LUCIE PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-335 AND 50-389

1.0 INTRODUCTION

On September 27, 1985, the staff issued Generic Letter (GL) 85-19, "Reporting Requirements on Primary Coolant Iodine Spikes." In GL 85-19 the staff recommended that the primary coolant iodine spike reporting requirements could be reduced from a short-term report (such as Special Report or Licensee Event Report) to an item which is included in the Annual Report. In addition, GL 85-19 also recommended that the existing requirements to shut down a plant if coolant iodine activity limits are exceeded for 800 hours in a 12-month period could be eliminated. By letter dated February 21, 1989, Florida Power & Light Company (the licensee) proposed to revise the St. Lucie Plant, Units 1 and 2 Technical Specifications (TS) Section 3.4.8 and the bases for these TS to conform to the recommendations of GL 85-19.

2.0 EVALUATION

The St. Lucie Plant, Units 1 and 2 TS currently include ACTION requirements to be taken when the specific activity of the primary coolant exceeds certain limits. The licensee proposes to include this information in the Annual Report. This information is similar to that previously required in the Licensee Event Report, but has been changed to more clearly designate the results to be included from the specific activity analysis and to delete the information regarding fuel burnup by the core region.

In GL 85-19, the staff determined that the existing requirements to shut down a plant if coolant iodine activity limits are exceeded for 800 hours in a 12 month period can be eliminated. In the past decade the quality of nuclear fuel improved greatly and therefore the normal coolant iodine activity (absent iodine spiking) is well below the limit. Appropriate actions would be taken long before accumulating 800 hours above the iodine activity limit. Additionally, 10 CFR 50.72(b)(1)(ii) requires that the NRC be immediately notified of fuel cladding failures that exceed expected values or that are caused by unexpected factors. Proper fuel management by the licensees and the existing reporting requirements would preclude ever approaching the limit. For these reasons this TS limit is no longer considered necessary.

3.0 TECHNICAL FINDING

Based on the above evaluation, and because the changes proposed by the licensee are in conformance with the recommendations of GL 85-19, the staff finds the proposed amendment acceptable.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments relate to changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

5.0 CONCLUSION

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

Date: January 18, 1990

Principal Contributor: Jan A. Norris