UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 30, 2000

Mr. T. F. Plunkett
President - Nuclear Division
Florida Power and Light Company
P.O. Box 14000
Juno Beach, Florida 33408-0420

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS REGARDING PRESSURE-TEMPERATURE LIMITS AND COLD OVERPRESSURE MITIGATION SYSTEM REQUIREMENTS (TAC NOS. MA9500 AND MA9502)

Dear Mr. Plunkett:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 208 to Facility Operating License No. DPR-31 and Amendment No. 202 to Facility Operating License No. DPR-41 for the Turkey Point Plant, Units Nos. 3 and 4, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated July 7, 2000, as supplemented October 2 and 4, 2000. The pressure-temperature limits specified in TS 3.4.9.1 and Figures 3.4-2, 3.4-3 and 3.4-4 have been modified and the Cold Overpressure Mitigation System (COMS) requirements have been changed. The COMS is the Westinghouse version of the Low Temperature Overpressure Protection system.

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,

Kahtan N. Jabbour

Kahtan N. Jabbour, Senior Project Manager, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-250 and 50-251

Enclosures:
1. Amendment No. 208 to DPR-31
2. Amendment No. 202 to DPR-41
3. Safety Evaluation

cc w/enclosures: See next page

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NRR-058

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 208
License No. DPR-31

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated July 7, 2000, as supplemented October 2 and October 4, 2000, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-31 is hereby amended to read as follows:

   (B) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 208, are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

2. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 30, 2000

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 202
License No. DPR-41

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

   A. The application for amendment by Florida Power and Light Company (the licensee) dated July 7, 2000, as supplemented October 2 and October 4, 2000, 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, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-41 is hereby amended to read as follows:

   (B) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 202, are hereby incorporated in the license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into the license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Richard P. Correia

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 30, 2000

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 208 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 202 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NOS. 50-250 AND 50-251

Replace the following pages of the Appendix A Technical Specifications with the attached pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

Remove pages	Insert pages
INDEX viii	INDEX viii
3/4 4-30	3/4 4-30
3/4 4-31	3/4 4-31
3/4 4-32	3/4 4-32
3/4 4-33	3/4 4-33
3/4 4-36	3/4 4-36
3/4 4-37	3/4 4-37
B3/4 4-7	B3/4 4-7
B3/4 4-8	B3/4 4-8
B3/4 4-9	B3/4 4-9
B3/4 4-12	B3/4 4-12
B3/4 4-15	B3/4 4-15

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3/4.4.9 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.9.1 The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4-2 and 3.4-3 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period, and

c. A maximum temperature change of less than or equal to 5°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the RCS T_{avg} and pressure to less than 200°F and 500 psig, respectively, within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.1.1 The Reactor Coolant System temperature and pressure shall be determined to be within the limits at least once per 30 minutes during system heatup, cooldown, and inservice leak and hydrostatic testing operations.

4.4.9.1.2 The reactor vessel material irradiation surveillance specimens shall be removed and examined, to determine changes in material properties, as required by 10 CFR Part 50, Appendix H. The results of these examinations shall be used to update Figures 3.4-2 and 3.4-3.

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate/Lower Shell Circumferential Weld Seams (Ht. #71249)

LIMITING ART VALUES AT 32 EFPY: 1/4T, 262°F
3/4T, 218°F

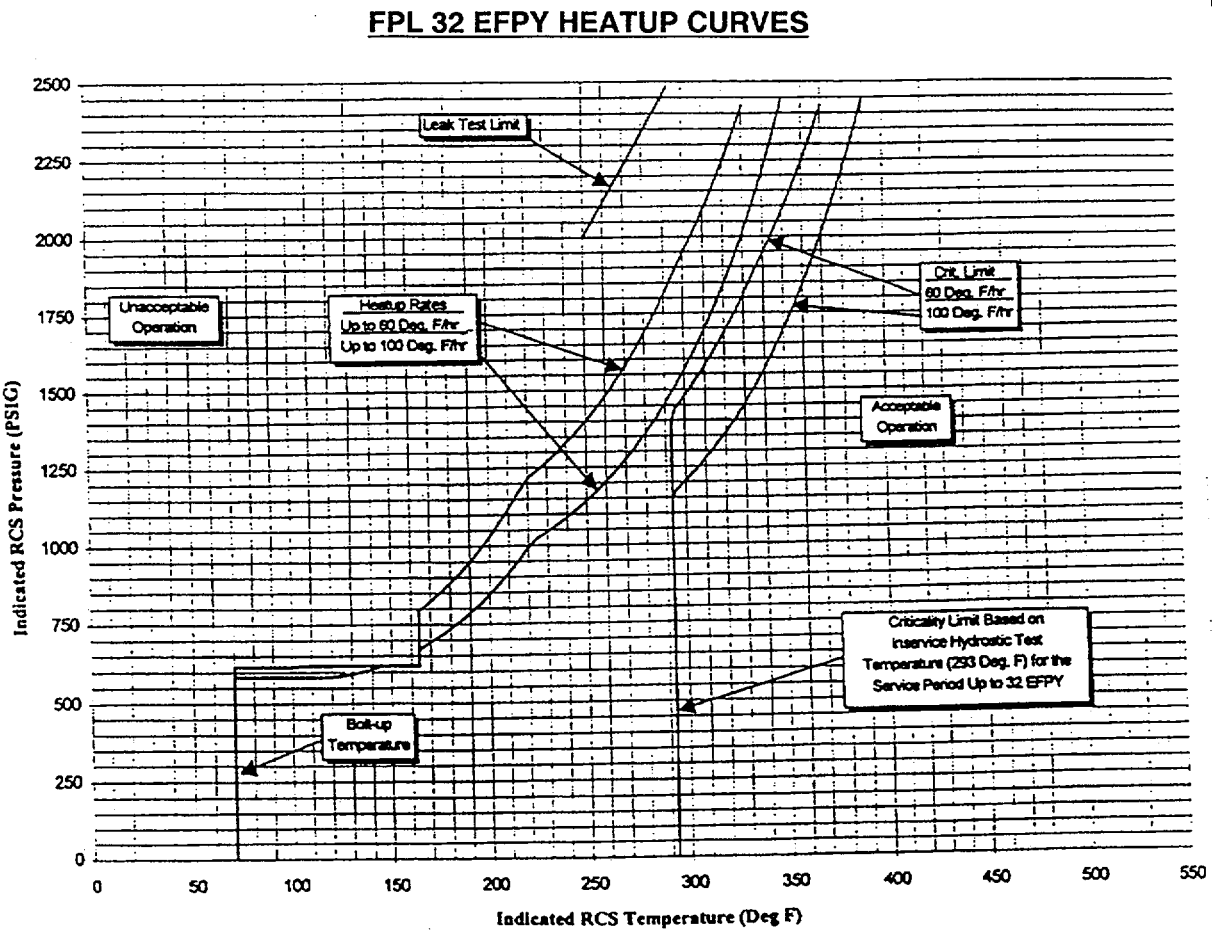

FIGURE 3.4-2 Turkey Point Units 3 and 4 Reactor Coolant System Heatup Limitations (Heatup Rate of 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

LIMITING MATERIAL: Intermediate/Lower Shell Circumferential Weld Seams (Ht. #71249)

LIMITING ART VALUES AT 32 EFPY: 1/4T, 262°F
3/4T, 218°F

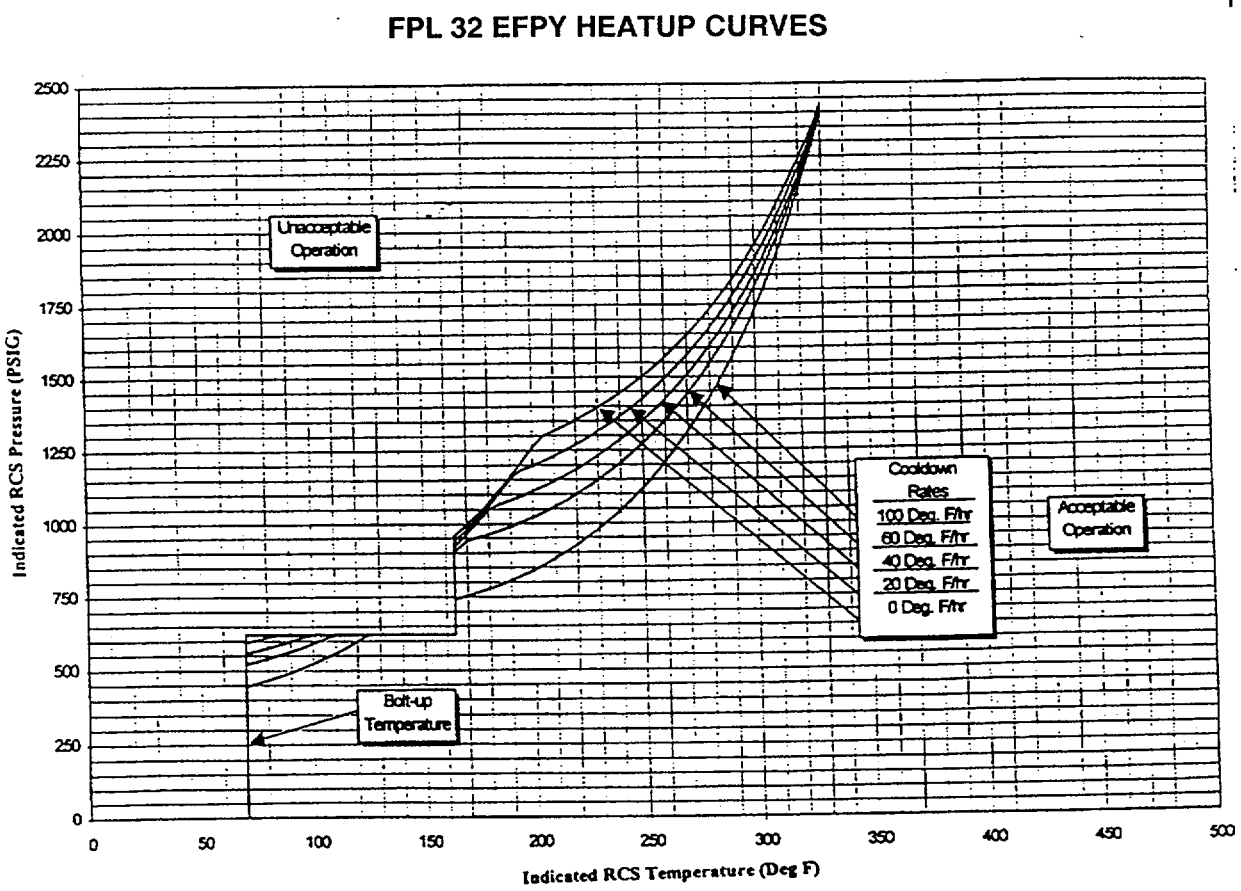

FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rate of 0, 20, 40, 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

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LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and below an RCS average coolant temperature of 275°F at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

a. Two power-operated relief valves (PORVs) with a lift setting of ≤ 468 psig, or

b. The RCS depressurized with a RCS vent of greater than or equal to 2.20 square inches.

APPLICABILITY MODES 4 (below an RCS average coolant temperature of 275°F), 5, and 6 with the reactor vessel head on.

ACTION:

a. With the high pressure safety injection flow paths to the RCS unisolated, restore isolation of these flow paths within 4 hours.

b. With one PORV inoperable in MODE 4 (below an RCS average coolant temperature of 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

c. With one PORV inoperable in Modes 5 or 6 with the reactor vessel head on, either (1) restore the inoperable PORV to OPERABLE status within 24 hours, or (2) complete depressurization and venting of the RCS through at least a 2.20 square inch vent within a total of 32 hours, or (3) complete depressurization and venting of the RCS through at least one open PORV and associated block valve within a total of 32 hours.

d. With both PORVs inoperable, either restore one PORV to OPERABLE status or complete depressurization and venting of the RCS through at least a 2.20 square inch vent within 24 hours.

e. In the event either the PORVs or a 2.20 square inch vent is used to mitigate an RCS pressure transient, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 30 days. The report shall describe the circumstances initiating the transient, the effect of the PORVs or RCS vent(s) on the transient, and any corrective action necessary to prevent recurrence. A Special Report is not required when such a transient is the result of water injection into the RCS for test purposes with an open vent path.

f. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.9.3.1 Each PORV shall be demonstrated OPERABLE by:

a. Performance of an ANALOG CHANNEL OPERATIONAL TEST* on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required OPERABLE and at least once per 31 days thereafter when the PORV is required OPERABLE.

b. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months; and

c. Verifying the PORV block valve is open at least once per 72 hours when the PORV is being used for overpressure protection.

d. While the PORVs are required to be OPERABLE, the backup nitrogen supply shall be verified OPERABLE at least once per 24 hours.*

4.4.9.3.2 The 2.20 square inch vent shall be verified to be open at least once per 12 hours** when the vent(s) is being used for overpressure protection.

4.4.9.3.3 Verify the high pressure injection flow path to the RCS is isolated at least once per 24 hours by closed valves with power removed or by locked closed manual valves.

* Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ 275°F.

** Except when the vent pathway is provided with a valve which is locked, sealed, or otherwise secured in the open position, then verify these valves open at least once per 31 days.

SPECIFIC ACTIVITY (Continued)

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 reactor 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 reactor coolant will be detected in sufficient time to take corrective action. 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 RCS are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are induced by normal load transients, reactor trips and startup and shutdown operations. During RCS heatup and cooldown, the temperature and pressure changes must be limited to be consistent with design assumptions and to satisfy stress limits for brittle fracture.

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 temperature and pressure changes during heatup and cooldown are limited to be consistent with the requirements given in the ASME Boiler and Pressure Vessel Code, Section XI, Appendix G:

PRESSURE/TEMPERATURE LIMITS (Continued)

1. The reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures 3.4-2 and 3.4-3 for the service period specified thereon:

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation; and

   b. Figures 3.4-2 and 3.4-3 define limits to assure prevention of non-ductile failure only. For normal operation, other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity, may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. These limit lines shall be calculated periodically using methods provided below,

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the steam generator is below 70°F,

4. The pressurizer heatup and cooldown rates shall not exceed 100°F/h and 200°F/h, respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F, and

5. System preservice hydrotests and inservice leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with the NRC Standard Review Plan, the version of the ASTM E185 standard required by 10 CFR 50, Appendix H, and in accordance with additional reactor vessel requirements.

The properties are then evaluated in accordance with Appendix G of the 1996 Edition of Section XI of the ASME Boiler and Pressure Vessel Code and the additional requirements of 10 CFR 50, Appendix G and the calculation methods described in Westinghouse Report WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{NDT}, at the end of 32 effective full power years (EFPY) of service life. The 32 EFPY service life period is chosen such that the limiting RT_{NDT}, at the 1/4T location in

PRESSURE/TEMPERATURE LIMITS (Continued)

the core region is greater than the RT_{NDT}, of the limiting unirradiated material. The selection of such a limiting RT_{NDT} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The heatup and cooldown limit curves, Figures 3.4-2 and 3.4-3 are composite curves prepared by determining the most conservative case with either the inside or outside wall controlling, for any heatup rate up to 100 degrees F per hour and cooldown rates of up to 100 degrees F per hour. The heatup and cooldown curves were prepared based upon the most limiting value of predicted adjusted reference temperature at the end of the applicable service period (32 EFPY).

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests are shown in Tables B 3/4.4-1 and B 3/4.4-2. Reactor operation and resultant fast neutron (E greater than 1 MeV) irradiation can cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence and chemistry factors of the material has been predicted using Regulatory Guide 1.99, Revision 2, dated May 1988, "Radiation Embrittlement of Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period.

The actual shifts in RT_{NDT}, of the vessel materials will be established periodically during operation by removing and evaluating, in accordance with the version of the ASTM E185 standard required by 10 CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are essentially identical, the measured transition shift for a sample can be applied with confidence to the adjacent section of the reactor vessel.

Since the limiting beltline materials (Intermediate to Lower Shell Circumferential Weld) in Units 3 and 4 are identical, the RV surveillance program was integrated and the results from capsule testing is applied to both Units. All available surveillance capsule results for the Unit 3 and 4 reactor vessel were used with the methodology in Regulatory Guide 1.99, Revision 2, to provide

PRESSURE/TEMPERATURE LIMITS (Continued)

limiting material properties information for generating the heatup and cooldown curves in Figures 3.4-2 and 3.4-3. The integrated surveillance program along with similar identical reactor vessel design and operating characteristics allows the same heatup and cooldown limit curves to be applicable at both Unit 3 and Unit 4.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section XI of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10 CFR Part 50.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semielliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section XI as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against nonductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil-ductility reference temperature, RT_{NDT}, is used and this includes the radiation-induced shift, ΔRT_{NDT}, corresponding to the end of the period for which heatup and cooldown curves are generated.

The ASME approach for calculating the allowable limit curves for various heatup and cooldown rates specifies that the total stress intensity factor, K_I, for the combined thermal and pressure stresses at any time during heatup or cooldown cannot be greater than the reference stress intensity factor, K_{IR}, for the metal temperature at that time. K_{IR} is obtained from the reference fracture toughness curve, defined in Appendix G to the ASME Code. The K_{IR} curve is given by the equation:

$$K_{IR} = 26.78 + 1.223 \exp [0.0145(T-RT_{NDT} + 160)] \quad (1)$$

Where: K_{IR} is the reference stress intensity factor as a function of the metal temperature T and the metal nil-ductility reference temperature RT_{NDT}. Thus, the governing equation for the heatup-cooldown analysis is defined in Appendix G of the ASME Code as follows:

$$C\, K_{IM} + K_{IT} \leq K_{IR} \quad (2)$$

PRESSURE/TEMPERATURE LIMITS (Continued)

Finally, the 10 CFR 50 Appendix G rule which addresses the metal temperature of the closure head flange and vessel flange regions is considered. The rule states that the minimum metal temperature for the flange regions should be at least 120°F higher than the limiting RT_{NDT} for these regions when the pressure exceeds 20 percent of the preservice hydrostatic test pressure (621 psig). Since the limiting RT_{NDT} for the flange regions for Turkey Point Units 3 and 4 is 44 F, the minimum temperature required for pressure of 621 psig and greater based on the Appendix G rule is 164°F. The heatup and cooldown curves as shown in Figures 3.4-2 and 3.4-3 clearly satisfy the above requirement by ample margins.

Finally, the composite curves for the heatup rate data and the cooldown rate data are adjusted for possible errors in the pressure and temperature sensing instruments by the values indicated on the respective curves.

The limitations imposed on the pressurizer heatup and cooldown rates and spray water temperature differential are provided to assure that the pressurizer is operated within the design criteria assumed for the fatigue analysis performed in accordance with the ASME Code requirements.

OVERPRESSURE MITIGATING SYSTEM

The Technical Specifications provide requirements to isolate High Pressure Safety Injection from the RCS and to prevent the start of an idle RCP if secondary temperature is more than 50 F above the RCS cold leg temperatures. These requirements are designed to ensure that mass and heat input transients more severe than those assumed in the low temperature overpressurization protection analysis cannot occur.

The OPERABILITY of two PORVs or an RCS vent opening of at least 2.20 square inches ensures that the RCS will be protected from pressure transients which could exceed the limits of Appendix G to 10 CFR Part 50 when one or more of the RCS cold legs are less than or equal to 275°F. Either PORV has adequate relieving capability to protect the RCS from overpressurization when the transient is limited to either: (1) the start of an idle RCP with the secondary water temperature of the steam generator less than or equal to 50°F above the RCS cold leg temperatures including margin for instrument error, or (2) the start of a HPSI pump and its injection into a water-solid RCS. When the PORVs or 2.2 square inch area vent is used to mitigate a plant transient, a special report is submitted. However, minor increases in pressure resulting from planned plant actions, which are relieved by designated openings in the system, need not be reported. The Overpressure Mitigating System setpoint includes an allowance for instrument uncertainty.

REACTOR MATERIAL SURVEILLANCE PROGRAM

Each Type I capsule contains 28 V-notch specimens, ten Charpy specimens machined from each of the two shell forgings. The remaining eight Charpy specimens are machined from correlated monitor material. In addition, each

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO.208 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 202 TO FACILITY OPERATING LICENSE NO. DPR-41

FLORIDA POWER AND LIGHT COMPANY

TURKEY POINT UNIT NOS. 3 AND 4

DOCKET NOS. 50-250 AND 50-251

1.0 INTRODUCTION

By letter dated July 7, 2000, as supplemented October 2 and 4, 2000, Florida Power & Light Company (the licensee) submitted proposed changes to the Technical Specifications (TS) for the Turkey Point Plant, Units 3 and 4. The changes are related to the pressure-temperature (P/T) limit curves, and the Cold Overpressure Mitigation System (COMS) requirements. The COMS is the Westinghouse version of the Low Temperature Overpressure Protection (LTOP) system. The supplemental letter dated October 2 and 4, 2000, provided clarifying information that did not affect the proposed no significant hazards consideration determination.

Specifically, the licensee has proposed the following changes:

Revise TS Figures 3.4-2 and 3.4-3 and delete TS Figure 3.4-4 to reflect the proposed changes to the PT limits.

Revise TS 3.4.9.3, "Overpressure Mitigation System," by changing the relief valve setting.

Add a footnote to the TS 4.4.9.3.1a and TS 4.4.9.3.1d surveillance requirements for the overpressure mitigation system regarding surveillance completion and change the text in TS 4.4.9.3.1d from "backup air supply" to "backup nitrogen supply."

Revise the titles of Figures 3.4-2 and 3.4-3, and delete two references to Figure 3.4-4 in the TS Index and in TS 3/4.4.9, "Pressure/Temperature Limits." These changes are administrative in nature. Also, change the TS bases regarding the P/T limits to make them consistent with the TS changes.

2.0 EVALUATION

2.1 P/T Limits and T_{enable}

2.1.1 Background

The licensee proposed revisions to the P/T limits and T_{enable} to provide new limits that are valid to 32 effective full power years (EFPY). The proposed changes are based, in part, on the use of American Society of Mechanical Engineers (ASME) Code Cases N-588 and N-641, which were reviewed by the staff. An exemption for use of these ASME Code Cases for Turkey Point Units 3 and 4 has been addressed in a separate action.

The U.S. Nuclear Regulatory Commission (NRC) has established requirements in *Title 10, Code of Federal Regulations* (10 CFR), Part 50, to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the P/T limit curves and T_{enable} based on the following NRC regulations and guidance: 10 CFR 50, Appendix G; Generic Letter (GL) 88-11; GL 92-01, Rev. 1; GL 92-01, Rev. 1, Supplement 1; Regulatory Guide (RG) 1.99, Rev. 2; and Standard Review Plan (SRP) Sections 5.2.2 and 5.3.2. GL 88-11 advised licensees that the staff would use RG 1.99, Rev. 2, to review P/T limit curves. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation. GL 92-01, Rev. 1, requested that licensees submit their reactor pressure vessel (RPV) data for their plants to the staff for review. GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. These data are used by the staff as the basis for the staff's review of P/T limit curves and T_{enable}, and as the basis for the staff's review of pressurized thermal shock (PTS) assessments (10 CFR 50.61 assessments). Appendix G to 10 CFR Part 50 requires that P/T limit curves and T_{enable} be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the ASME Boiler and Pressure Vessel Code. Appendix G to 10 CFR Part 50 also provides minimum temperature requirements that must be considered in the development of the P/T limit curves.

SRP Section 5.3.2 provides an acceptable method of determining the P/T limit curves for ferritic materials in the beltline of the RPV based on the linear elastic fracture mechanics methodology of Appendix G to Section XI of the ASME Code. The basic parameter of this methodology is the stress intensity factor K_I, which is a function of the stress state and flaw configuration. Appendix G requires a safety factor of 2.0 on stress intensities resulting from reactor pressure during normal and transient operating conditions, and a safety factor of 1.5 for hydrostatic testing curves. The methods of Appendix G postulate the existence of a sharp surface flaw in the RPV that is normal to the direction of the maximum stress. This flaw is postulated to have a depth that is equal to 1/4 of the RPV beltline thickness and a length equal to 1.5 times the RPV beltline thickness. The critical locations in the RPV beltline region for calculating heatup and cooldown P/T curves are the 1/4 thickness (1/4T) and 3/4 thickness (3/4T) locations, which correspond to the maximum depth of the postulated inside surface and outside surface defects, respectively.

The Appendix G ASME Code methodology requires that licensees determine the adjusted reference temperature (ART or adjusted RT_{NDT}). The ART is defined as the sum of the initial

(unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation (ΔRT_{NDT}), and a margin (M) term.

The ΔRT_{NDT} is a product of a chemistry factor and a fluence factor. The chemistry factor is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Rev. 2, or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The margin term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the chemistry factor was determined using the tables in RG 1.99, Rev. 2, or surveillance data. The margin term is used to account for uncertainties in the values of the initial RT_{NDT}, the copper and nickel contents, the fluence and the calculational procedures. RG 1.99, Rev. 2, describes the methodology to be used in calculating the margin term.

2.1.2 Licensee Evaluation

The licensee submitted P/T limit curves, LTOP setpoints and T_{enable} valid for operations up to 32 EFPY. As described in Westinghouse Electric Company LLC report WCAP-15092, Rev. 3, "Turkey Point Units 3 and 4 WOG Reactor Vessel 60-Year Evaluation Minigroup Heatup and Cooldown Limit Curves for Normal Operation" (dated May 2000), similarities in construction, materials and operation of the two units allow the licensee to consider the two units together, with the development of P/T limit curves, LTOP setpoints and T_{enable} valid for both units. It should be noted that although the report addresses conditions for both 32 and 48 EFPY operating periods, the license amendment requested by the licensee is only for the 32 EFPY period.

From WCAP-15092, Rev. 3, the licensee determined that the limiting ART for the RPV of each unit is from the intermediate to lower shell circumferential weld, heat number 71249. The other materials considered in this report are the intermediate and lower shell forgings of each RPV. Results from the Turkey Point Units 3 and 4 surveillance programs are available for the four forging heats, with a total of four data points for the limiting weld, heat number 71249.

For the forgings, two data points are available for each forging, except for the intermediate shell forging of Unit 4, for which only a single data point is available. For the forgings with multiple data points, the licensee found that the available data fulfilled the credibility requirements of RG 1.99, Rev. 2, and hence the data could be used to develop a heat-specific chemistry factor, and a reduced margin term, for use in evaluating the ART for these materials, in accordance with Regulatory Position 2.1 of the RG. For the intermediate shell forging of unit 4, the chemistry factor was determined from Table 2 of RG 1.99, Rev. 2, and the full margin term was used, in accordance with Regulatory Position 1.1 of the RG.

For the limiting RPV material, weld heat number 71249, the licensee evaluated the four available surveillance data points for the heat. Because the surveillance weld materials had a higher copper concentration than the RPV welds, the licensee's analysis of the surveillance data did not incorporate a chemistry factor ratio adjustment as outlined in Regulatory Position 2.1 of the RG. Evaluation of these data in WCAP-15092, Rev. 3, indicated that the surveillance data did not satisfy the credibility requirements of RG 1.99, Rev. 2, and hence the chemistry factor for weld heat number 71249 was determined from Table 1 of RG 1.99, Rev. 2, and the full margin term was used, in accordance with Regulatory Position 1.1 of the RG.

A summary of the chemistry factors for each heat from the two RPVs is provided in Table 1. A summary of the ART values determined by the licensee for the 1/4T and 3/4T locations is provided in Tables 2 and 3.

From the data in Tables 2 and 3, the licensee evaluated P/T limits using the methodology provided in ASME Code Case N-588. Since this Code Case requires postulation of both axial and circumferential flaws, the limiting 1/4T ARTs for axial and circumferential flaws are 124.3°F and 262.2°F, respectively, and the limiting 3/4T ARTs for axial and circumferential flaws are 116.6°F and 218.3°F, respectively. The P/T limits provided by the licensee do not provide margins for instrument errors.

For the T_{enable}, ASME Code Case N-641 indicates that the LTOP system should be effective below RT_{NDT} + 40°F for axial surface flaws (e.g., for axial welds and base materials) and RT_{NDT} -85°F for circumferential surface flaws (e.g., for circumferential welds). From the data in Tables 2 and 3, the licensee determined that a T_{enable} defined by 262°F - 85°F + 24.6°F, or 201.6°F, is justified for Turkey Point, where the factor of 24.6°F represents the temperature lag between the RPV 1/4T location and the monitored reactor coolant bulk temperature. However, the licensee will use a conservative T_{enable} of 275°F, which is the current T_{enable}.

Table 1: Comparison of Chemistry Factors for Turkey Point Units 3 and 4

Material	Chemical Composition		Chemistry Factor	
	Copper (wt. %)	Nickel (wt. %)	Position 1.1	Position 2.1
Unit 3				
Intermediate shell forging (123P461VA1)	0.06	0.70	37.0	14.6[a]
Lower shell forging (123S266VA1)	0.08	0.67	51.0	42.7[a]
Unit 4				
Intermediate shell forging (123P481VA1)	0.05	0.68	31.0[a]	[b]
Lower shell forging (122S180VA1)	0.06	0.74	37.0	5.4[a]
Both Units				
Intermediate to lower shelf circumferential weld (71249)	0.23	0.59	167.6[a]	184.3[c]

[a] Value used by licensee.
[b] Insufficient data for a determination.
[c] Data do not satisfy credibility criteria of RG 1.99, Rev. 2.

Table 2: Comparison of ART Values for Turkey Point Units 3 and 4 for the 1/4T Location

Material	1/4T Fluence (10^{19} n/cm^2)	Chem. Factor (°F)	ΔRT_{NDT} (°F)	Init. RT_{NDT} (°F)	Margin (°F)	ART (°F)
Unit 3						
Int. Forging	3.11	14.6	18.9	40	17	75.9
Lower Forging	2.49	42.7	53.2	30	17	100.2
Unit 4						
Int. Forging	3.11	31.0	40.3	50	34	124.3
Lower Forging	2.49	5.4	6.7	40	17	63.7
Both Units						
Weld	1.87	167.6	196.2	10	56	262.2

Table 3: Comparison of ART Values for Turkey Point Units 3 and 4 for the 3/4T Location

Material	3/4T Fluence (10^{19} n/cm^2)	Chem. Factor (°F)	ΔRT_{NDT} (°F)	Init. RT_{NDT} (°F)	Margin (°F)	ART (°F)
Unit 3						
Int. Forging	1.20	14.6	15.3	40	17	72.3
Lower Forging	0.96	42.7	42.3	30	17	89.3
Unit 4						
Int. Forging	1.20	31.0	32.6	50	34	116.6
Lower Forging	0.96	5.4	5.3	40	17	62.3
Both Units						
Weld	0.72	167.6	152.3	10	56	218.3

2.1.3 **Staff Evaluation**

The staff performed an independent calculation of the ART values for the limiting material using the methodology in RG 1.99, Rev. 2, and the data for the beltline materials provided in the licensee's submittal is consistent with that in the NRC's reactor vessel integrity database (RVID). Based on this review, the staff verified the licensee's limiting axial and circumferential materials for Turkey Point Units 3 and 4 reactor vessel, along with the ARTs for each orientation at both the 1/4T and 3/4T locations.

In its evaluation of the surveillance data for the circumferential weld heat number 71249, the staff determined (as did the licensee) that the surveillance data do not meet the credible data requirements of RG 1.99, Rev. 2. However, the staff notes that for an evaluation of the data to be consistent with the guidance of RG 1.99, Rev. 2, the chemistry factor ratio adjustment described in Regulatory Position 2.1 should be performed on the data. This adjustment is necessary to ensure an appropriate consideration of the data and to provide an accurate assessment of the data. Considering the weld surveillance data with this chemistry factor ratio adjustment, the staff calculated a surveillance chemistry factor of 155.4°F in accordance with Regulatory Position 2.1 of RG 1.99, Rev. 2. This value is lower than the value determined by the licensee (184.3°F) and lower than the chemistry factor for Regulatory Position 1.1 of RG 1.99, Rev. 2 (167.55°F). As described previously, the staff determined that the surveillance data are not credible in accordance with RG 1.99, Rev. 2, and therefore the chemistry factor for the RPV weld should be 167.55°F in accordance with Regulatory Position 1.1 of RG 1.99, Rev. 2.

In a related matter, the staff notes that the RVID information for the Turkey Point RPVs lists the circumferential weld (heat number 72442) between the nozzle belt and the intermediate shell as exhibiting a relatively high RT_{PTS} at end of life, although the neutron fluence (~0.3 x 10^{19} n/cm^2)

is an order of magnitude less than that for the materials considered by the licensee. Although this material is not the limiting material for Turkey Point, future additions to surveillance data or changes to embrittlement correlations could result in this material becoming a more significant consideration in determining the limiting material, and as such the licensee is encouraged to track this material and consider it in future submittals.

The staff performed check calculations to verify the P/T limit curves using the appropriate limiting ART values for the Turkey Point Units 3 and 4 RPV beltline materials. The staff's calculations assumed steady-state conditions throughout the entire heatup or cooldown curve, whereby a constant thermal stress intensity (K_{It}) and a constant temperature difference between the reactor coolant bulk and the vessel thickness were assumed at all points on each curve. From these calculations, the staff found moderate differences in cooldown and heatup curves at low coolant temperatures and using high cooldown and heatup rates (100°F/hr). These differences were found to diminish with reduced cooldown or heatup rate, and ultimately disappeared with calculations assuming isothermal conditions. After discussion with the licensee regarding the time-dependent thermal and stress history used in the licensee's calculations, the licensee provided the staff with water temperature, RPV temperature and factor K_{It} data for the 100°F/hr heatup and cooldown curves. Using the licensee supplied data, excellent agreement was found between the staff and licensee calculations, verifying the appropriateness of the curves submitted by the licensee. The staff also found that the minimum temperature requirements of Table 1 of Appendix G to 10 CFR Part 50 were properly implemented in the P/T limit curves.

The licensee clearly indicated that the P/T limit curves do not provide margins to account for instrument errors, which in general should be accounted for in the curves. Since the licensee used the ASME K_{Ia} fracture toughness curve for determination of the P/T limit curves, in lieu of the ASME K_{Ic} fracture toughness curve permitted by ASME Code Cases N-588 and N-641, the inherent conservative margins of the ASME K_{Ia} fracture toughness curve makes the imposition of additional margins to account for instrument error unnecessary. Therefore, these curves are acceptable.

The staff determined that the T_{enable} proposed by the licensee, 275°F, is more conservative than that permitted by ASME Code Case N-641.

Thus, the staff determined that the P/T limit curves and T_{enable} satisfy the requirements in Paragraph IV.A.2 of Appendix G of 10 CFR Part 50 as modified by Code Cases N-588 and N-641, and hence, the requirements of 10 CFR 50.60.

2.1.4 Summary

The staff concludes that the proposed P/T limit curves and T_{enable} for heatup and cooldown of the reactor coolant system satisfy the requirements in Appendix G to Section XI of the ASME Code, as modified by Code Cases N-588 and N-641, and Appendix G of 10 CFR 50, for 32 EFPY. The proposed P/T limit curves and T_{enable} also satisfy GL 88-11, because the method in RG 1.99, Rev. 2, was used to calculate the ART. Hence, the proposed P/T limit curves and T_{enable} are acceptable for incorporation in the Turkey Point Units 3 and 4 TS.

2.2 LTOP

2.2.1 Background

The proposed TS amendments would also modify the COMS setpoint in accordance with the amended P/T limits. The licensee proposed to amend surveillance requirements to delay the operability verification of the COMS Analog Channel Operation Test (ACOT) and the backup nitrogen supply until 12 hours after decreasing the reactor coolant system (RCS) cold leg temperature below the COMS enable temperature.

The COMS uses the pressurizer power-operated relief valves (PORV) for LTOP during startup or shutdown operations, and supplements the normal plant operational administrative controls and the water relief valves in the residual heat removal system (RHRS) when they are unavailable or inadvertently isolated from the RCS. By choosing appropriate setpoints, the COMS is designed with the capability to automatically prevent the RCS pressure from exceeding the P/T limits, and prevent the reactor vessel from being exposed to conditions of fast propagating brittle fracture. Once the system is enabled, no operator action is involved for the COMS to perform its intended pressure mitigation function.

Modifications of the COMS setpoints for the PORVs as a result of changes to the P/T limits to extend the reactor operation period are routinely made on nuclear plants.

2.2.2 Staff Evaluation

2.2.2a COMS PORV Setpoints

The current LCO [limiting condition for operation] 3.4.9.3 specifies that, in MODE 4 (below an RCS average coolant temperature of 275 °F), MODE 5, and MODE 6 with the reactor vessel head on, the high pressure safety injection (HPSI) flow paths to the RCS shall be isolated, and the COMS shall be operable by having either (a) two PORVs with a lift setting of 415 ± 15 psig, or (b) the RCS depressurized with an RCS vent of ≥ 2.20 square inches. The proposed TS changes would revise the PORV lift setting from 415 ± 15 psig to ≤ 468 psig.

The revised PORV lift setpoint of ≤ 468 psig for the COMS operation is the analytical limit derived from an evaluation of the revised P/T limit curves and applicable design basis overpressure events, and is adjusted to account for instrument uncertainty.

In Enclosure 2 (see Reference 4) to the July 7, 2000, letter, the licensee provided an LTOP report documenting the development of the COMS PORV setpoints for Turkey Point for operation up to 32 EFPY. These setpoints were developed using the Westinghouse methodology outlined in WCAP-14040-NP-A, Rev. 2 (Reference 5), which has been approved by the NRC for Westinghouse-designed PWRs, and is appropriate for use at Turkey Point Units 3 and 4. The setpoint methodology considers the limiting Appendix G pressure limits, adjustments for pressure measurement uncertainties, and the PORV piping/structural analysis limitations.

The COMS setpoint methodology considers both the mass-addition and heat-addition LTOP design basis transients, which are assumed to occur when the RCS is water solid. Each of

these transients assumes as an initial condition that the RHRS is isolated from the RCS, and thus the relief capability of the RHRS valves is not available. For each transient, calculations are made for pressure overshoots during the delay time before the valve starts to move and during the time the valve is moving to the full open position.

The mass-addition transient assumes mass injection into a water solid RCS from one HPSI pump. The assumption of inadvertent actuation of one HPSI pump as the limiting mass addition transient is acceptable because LCO 3.4.9.3 requires the HPSI injection flow paths to the RCS to be isolated. The limiting heat addition transient is assumed to be the startup of an idle reactor coolant pump with the secondary water temperature of the steam generators 50°F above the RCS cold leg temperature and the inadvertent isolation of the RHR system. This results in a sudden heat input to a water solid RCS from the steam generators, creating an increasing RCS pressure transient. The assumption of the RCS temperature no more than 50°F lower than the SG secondary side temperature is acceptable because of the provisions of LCO 3.4.1.3, and 3.4.1.4.1 preventing the start of an idle RCP if secondary temperature is more than 50°F above the RCS cold leg temperatures.

Of the two transients, the licensee determined that the mass-addition transient is limiting for defining the COMS setpoint for Turkey Point. This conclusion is based on a plant-specific analysis (Reference 6) of 0 percent steam generator tube plugging (SGTP) performed by Westinghouse, which demonstrated that, at the low RCS temperatures where the limiting setpoint is determined, pressure overshoots during the mass injection transients far exceed pressure overshoots during the heat addition events. Therefore, COMS setpoints are based on the mass injection transient.

Section 3 of the COMS setpoint report (Reference 4) documents the major assumptions used in the COMS setpoint development. The assumptions include a reactor vessel minimum bolt-up temperature of 70 °F, SGTP of 0 and 20 percent, a single failure assumption of one PORV failed closed, the PORV stroke open and close times of 3.05 and 2 seconds, respectively, a process delay of 0.3 seconds, and plant specific instrumentation uncertainties. The setpoints conservatively account for the pressure measurement uncertainty of 70 psi associated with the wide range pressure transmitter, and a pressure differential of 57 psi between the wide range transmitter and the reactor vessel limiting beltline region. In addition to the Appendix G limits, an 800 psig pressure limit is incorporated to address pressurizer PORV pipe loading considerations.

Section 5 of the COMS setpoint report provides the results of the COMS setpoint development. In Reference 7, it was determined that the pressure overshoots for the 20 percent SGTP condition would increase by approximately 4 psi over 0 percent SGTP. Table 1 of the setpoint report tabulates the mass injection overshoots for 0 percent SGTP case, with an adjustment of 4 psi for the 20 percent SGTP case. Table 2 tabulates the derivation of the Appendix G limits as a function of RCS temperature for 32 EFPY. The Appendix G limits are derived consistent with the methodology of WCAP-14040-NP-A in that the steady state limits of the RCS P/T limit curves are used with relaxation allowed by ASME Code Case N-514, the reduction to account for 70 psi pressure measurement uncertainty and 57 psi differential pressure between the transmitter and beltline region, and the PORV piping loading limitation of 800 psig. Based on the Appendix G limit results, the limiting overshoots of the mass addition transient shown in Table 1, and additional pressure overshoot adjustment for 20 percent SGTP, the licensee found

that the maximum allowable PORV setpoint at 32 EFPY is 468 psig. The staff has reviewed the PORV setpoint calculation and finds the result acceptable.

The proposed TS change to revise the PORV lift setting to ≤ 468 psig would provide low temperature overpressure protection such that the Appendix G P/T limits will not be exceeded. Therefore, the change is acceptable.

2.2.2b COMS Surveillance Requirements

Current Turkey Point TS Surveillance Requirement (SR) 4.4.9.3.1a requires the performance of an analog channel operational test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required to be OPERABLE and at least once per 31 days thereafter when the PORV is required to be OPERABLE. SR 4.4.9.3.1d requires that while the PORVs are required to be OPERABLE, the backup air supply shall be verified operable at least once per 24 hours. The licensee requested that a footnote be added to these SRs to state that "not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ 275 °F." The licensee also requested a change for SR 4.4.9.3.1d by replacing "backup air supply" with "backup nitrogen supply."

The proposed change to SR 4.4.9.3.1d from "backup air supply" to "backup nitrogen supply" is an administrative change which reflects the original and current plant configuration where nitrogen has always been used as opposed to the air mentioned in this SR. Therefore, it is acceptable.

The added footnote would allow for a delay of 12 hours for the completion of ACOT and verification of the PORV backup nitrogen supply operability. The licensee states that the need for a 12-hour window to complete the ACOT and the PORV backup nitrogen system operation verification is based on the timing sequence of the RHRS entry point of 350 °F and the COMS enable temperature of 275 °F. Extending the time period for performing these surveillances will allow the operators to focus on the transition from MODE 3 (hot standby) to MODE 4 (hot shutdown), and to stabilize the plant on RHR cooling prior to performing any COMS surveillance.

The licensee emphasizes the reliability of the Turkey Point instrument air system (IAS) in that each Turkey Point IAS has two air compressors, one diesel-driven and one motor-driven. The diesel-driven compressors are capable of supplying the required capacity without reliance on external power sources. To achieve a continuous reliable source of instrument air, a single air compressor is sized for the expected instrument air demand of both Units 3 and 4. Because the IAS is normally operated with the Unit 3 and 4 cross-connect valve open, the two systems will function as a common system. This arrangement provides up to 300 percent redundancy in the instrument air supply capacity. Therefore, the licensee contends that delaying completion of the channel operation test (COT) and PORV backup nitrogen operability demonstration for up to 12 hours will not pose a significant safety hazard due to the inherent reliability and redundancy of the IAS, and the unlikelihood of a low temperature overpressure event occurring concurrently with the loss of the highly redundant IAS in the 12-hour time frame.

The logic of adding the footnote for the delay of ACOT surveillance is consistent with the Westinghouse Standard TS (NUREG-1431) SR 3.4.12.8, which also contains a note not requiring performance of COT on each required PORV until 12 hours after entering the LTOP

modes, or decreasing RCS cold leg temperature to ≤ the COMS enable temperature. This 12-hour delay is needed because the COT cannot be performed until the plant is in the LTOP modes when the PORV lift setpoint can be reduced to the COMS setting. In addition, the COMS provides for LTOP a supplement to the RHRS water relief valve, which, if not isolated, would be available for LTOP. Because of the low probability of a low temperature overpressure event concurrent with an inadvertent isolation from the RCS of the RHRS water relief valves during the 12-hour period, the staff concludes that adding a footnote to allow for 12-hour delay in the completion of the COMS ACOT and backup nitrogen supply verification is acceptable.

2.2.3 **Summary**

The staff has evaluated the licensee's request to (1) revise the COMS setpoints specified in TS LCO 3.4.9.3 to ≤ 468 psig; (2) replace the PORV "backup air supply" with "backup nitrogen supply" in SR 4.4.9.3.1d; and (3) delay the completion of surveillance of ACOT and backup nitrogen supply until 12 hours after the RCS decreasing to within the COMS enable temperature. Based on its evaluation, the staff concludes these proposed TS revisions are acceptable.

2.3 Administrative and Bases Changes

2.3.1 **Administrative Changes**

The licensee proposed to revise the titles of Figures 3.4-2 and 3.4-3, and delete two references to Figure 3.4-4 in the TS index and in TS 3/4.4.9, "Pressure/Temperature Limits." These changes are administrative in nature and are consistent with the approved TS changes. Therefore they are acceptable.

2.3.2 **Bases Changes**

The licensee also proposed changes to the TS bases regarding P/T limits. These changes are consistent with the approved TS changes.

2.4 References

1. Letter from R. J. Hovey (FPL) to NRC, "Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, Proposed License Amendments, 'Revised Pressure/Temperature (P/T) Curves, and Cold Overpressure Mitigation System (COMS) Setpoints'", April 27, 2000.

2. Letter from R. J. Hovey (FPL) to NRC, "Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251, Resubmittal of Proposed License Amendments, 'Revised Pressure-Temperature (P/T) Curves, and Cold Overpressure Mitigation System (COMS) Setpoints'", July 7, 2000.

3. WCAP-15092, Rev. 3, "Turkey Point Units 3 and 4, WOG Reactor Vessel 60-Year Evaluation Minigroup, Heatup and Cooldown Limit Curves for Normal Operation," May 2000.

4. Letter from H. A. Sepp (Westinghouse Electric Company) to NRC, "Low Temperature Overpressure Protection System Setpoints, Florida Power and Light Company, 32 EFPY and 48 EFPY for Turkey Point, Units 3 and 4," June 29, 2000, CAW-00-1405 (Proprietary), Enclosure 2 to Reference 2.

5. WCAP-14040-NP-A, Rev. 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves," January 1996.

6. FD-CSA-265, "Evaluation of COMS Setpoints for an Increase in PORV Stroke Opening Time at Turkey Point Units 3 & 4," Final Report, January 20, 1989.

7. SE-ICAT (95)-223, Turkey Point Units 3 & 4 COMS Setpoint Evaluation, L. Sheffield, May 23, 1995, which transmits the results of calculation note CN-ICAT(94)-038, Rev. 0 and Rev. 1; and SE-ICAT(95)-341, IMP Data Error - Turkey Point Units 3 & 4 COMS Setpoint Evaluation at 20% Tube Plugging Conditions, L. Sheffield, August 18, 1995, which transmits the results of calculation note CN-ICAT(94)-038, Rev.3. Note, there was no Rev. 2 of CN-ICAT(94)-038.

3.0 STATE CONSULTATION

Based upon a letter dated March 8, 1991, from Mary E. Clark of the State of Florida, Department of Health and Rehabilitative Services, to Deborah A. Miller, Licensing Assistant, U.S. Nuclear Regulatory Commission, the State of Florida does not desire notification of issuance of license amendments.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that 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 that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (65 FR 48751). The supplemental information provided on October 2 and 4, 2000, provided clarifying information only and did not affect the proposed no significant hazards consideration determination. Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has 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, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: October 30, 2000

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