LICENSE AUTHORITY FILE C

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

DO NOT REMOVE

POSTED
Amdt. 128
to DPR-41

January 10, 1989

Docket Nos. 50-250
and 50-251

Mr. W. F. Conway
Senior Vice President-Nuclear
Nuclear Energy Department
Florida Power and Light Company
Post Office Box 14000
Juno Beach, Florida 33408-0420

Dear Mr. Conway:

SUBJECT: TURKEY POINT UNITS 3 AND 4 - ISSUANCE OF AMENDMENTS RE: PRESSURE AND TEMPERATURE (P/T) LIMITS (TAC NOS. 69390 AND 69391)

The Commission has issued the enclosed Amendment No. 134 to Facility Operating License No. DPR-31 and Amendment No. 128 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 transmitted by letter dated September 21, 1988.

These amendments revise Section 3.1.2 of the TS by incorporating modified P/T limits for the Reactor Coolant System (RCS) and pressurizer. These P/T limits take the form of parametric curves which define the permissible operating envelope during reactor heatup, cooldown, criticality, and inservice leak and hydrostatic testing. Because the P/T limits are based partly on the most limiting nil-ductility temperature for the reactor vessel, it is necessary to periodically revise the limits to account for improved data on the effects of irradiation and other factors on the nil-ductility temperature. The Turkey Point reactor pressure vessel surveillance program provides updated materials data for refining the estimated effects of irradiation on the nil-ductility temperature. The P/T limits currently in the TS are applicable up to 10 effective full power years (EFPY). These amendments replace the P/T curves with revised curves applicable up to 20 EFPY.

In addition to the proposed modification of P/T limits, the amendments also convert the TS to the standard format and revise the "Bases" section to be consistent with the revised P/T curves.

The cover letter of your application indicated that low temperature overpressure protection by your COMS system remained acceptable, and the PORV setpoint would remain at 415 psi under these conditions with the new curves. Our review of the curves, combined with reference to our earlier Safety Evaluation dated December 23, 1982, satisfied us that the P/T curves had not changed significantly for the heatup rate (0°F/hr.) and temperature (100°F) used as the bases for the PORV setpoint to provide low temperature overpressure protection.

A copy of the Safety Evaluation, including a Final Determination of No Significant Hazards Consideration, is enclosed. A copy of the Notice of Issuance is also enclosed. The Notice of Issuance will also be included in the Commission's biweekly Federal Register notice.

Sincerely,

Original signed by

Gordon E. Edison, Sr. Project Manager
Project Directorate II-2
Division of Reactor Projects-I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 134 to DPR-31
2. Amendment No. 128 to DPR-41
3. Safety Evaluation
4. Notice of Issuance

cc w/enclosures:
See next page

*See previous concurrences

LA:PDII-2*	PM:PDII-2 GEE	D:PDII-2	OGC-WF
DMiller	GEdison:bd	HBerkow	
11/17/88	01/10/89	01/10/89	01/~~9/89~~ M. 1/10/89
RSB:BC*	EMTB:BD		
WHodges	CYCheng		
11/18/88	01/10/89		

Mr. W. F. Conway
Florida Power and Light Company

Turkey Point Plant

cc:
Harold F. Reis, Esquire
Newman and Holtzinger, P.C.
1615 L Street, N.W.
Washington, DC 20036

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

John T. Butler, Esquire
Steel, Hector and Davis
4000 Southeast Financial
Center
Miami, Florida 33131-2398

Mr. J. Odom, Vice President
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

County Manager of Metropolitan
Dade County
Miami, Florida 33130

Resident Inspector
U.S. Nuclear Regulatory Commission
Turkey Point Nuclear Generating Station
Post Office Box 57-1185
Miami, Florida 33257-1185

Jacob Daniel Nash
Office of Radiation Control
Department of Health and
Rehabilitative Services
1317 Winewood Blvd.
Tallahassee, Florida 32399-0700

Intergovernmental Coordination
and Review
Office of Planning & Budget
Executive Office of the Governor
The Capitol Building
Tallahassee, Florida 32301

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

Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
Suite 2900
101 Marietta Street
Atlanta, Georgia 30323

Attorney General
Department of Legal Affairs
The Capitol
Tallahassee, Florida 32304

Plant Manager
Turkey Point Nuclear Plant
Florida Power and Light Company
P.O. Box 029100
Miami, Florida 33102

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT PLANT UNIT NO. 3

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 134
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 September 21, 1988, 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. 134, 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 the date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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 10, 1989

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

FLORIDA POWER AND LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT PLANT UNIT NO. 4

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 128
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 September 21, 1988, 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. 128, 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 the date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Herbert N. Berkow

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 10, 1989

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 134 FACILITY OPERATING LICENSE NO. DPR-31

AMENDMENT NO. 128 FACILITY OPERATING LICENSE NO. DPR-41

DOCKET NO. 50-250 AND 50-251

Revise Appendix A as follows:

Remove Pages	Insert Pages
v	v
vi	vi
3.1-2	3.1-2, 3.1-2a
3.1-2a	3.1-2b
Fig. 3.1-1a	Fig. 3.1-1a
Fig. 3.1-1b	Fig. 3.1-1b
Fig. 3.1-1c	Fig. 3.1-1c
Fig. 3.1-1d	-
Fig. 3.1-2c	-
Fig. 3.1-2d	-
B3.1-2	B3.1-2
--	B3.1-2a
--	B3.1-2b
--	B3.1-2c
--	B3.1-2d
B3.1-3	B3.1-3

LIST OF TABLES

LIST OF FIGURES

REACTOR COOLANT SYSTEM

3.1.2 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.2a The Reactor Coolant System (except the pressurizer) temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.1-1a, 3.1-1b and 3.1-1c for both Unit 3 and Unit 4 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.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.1.2.b The pressurizer temperature shall be limited to:

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

b. A maximum cooldown of 200°F in any 1-hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

MODERATOR TEMPERATURE COEFFICIENT

3.1.2.1 The moderator temperature coefficient (MTC) shall be:

a) Less positive than or equal to 5.0×10^{-5} $\Delta k/k/^{\circ}F$ for all rods withdrawn, beginning of the cycle life (BOL), hot zero THERMAL POWER (HZP) conditions; and

b) Less positive than or equal to 5.0×10^{-5} $\Delta k/k/^{\circ}F$ from HZP to 70% RATED THERMAL POWER condition; and

c) Less positive than or equal to 5.0×10^{-5} $\Delta k/k/^{\circ}F$ from 70% RATED THERMAL POWER decreasing linearly to less positive than or equal to 0 $\Delta k/k/^{\circ}F$ at 100% RATED THERMAL POWER condition; and

d) Less negative than -3.5×10^{-4} $\Delta k/k/^{\circ}F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.2.1a, b, and c - MODES 1 and 2* only**.
Specification 3.1.2.1d - MODES 1, 2, and 3 only**.

ACTION:

a) With the MTC more positive than the limits of Specifications 3.1.2.1a, b, or c above, operation in MODES 1 and 2 may proceed provided:

1) Control rod withdrawal limits are established and maintained sufficient to restore the MTC to less positive or equal to limits described in 3.1.2.1a, b, and c above within 24 hours or be in HOT STANDBY within the next 6 hours. These withdrawal limits shall be in addition to the insertion limits of specification 3.2.1,

2) The control rods are maintained within the withdrawal limits established above until a subsequent calculation verifies that the MTC has been restored to within its limit for the all rods withdrawn condition; and

3) A Special Report is prepared and submitted to the Commission pursuant to Specification 6.9.3, within 10 days, describing the value of the measured MTC, the interim control rod withdrawal limits, and the predicted average core burnup necessary for restoring the MTC to within its limit for the all rods withdrawn condition.

b) With the MTC more negative than the limit of Specification 3.1.2.1d above, be in HOT SHUTDOWN within 12 hours.

* With K_{eff} greater than or equal to 1.

** The above limits may be suspended during the performance of LOW POWER PHYSICS TESTS.

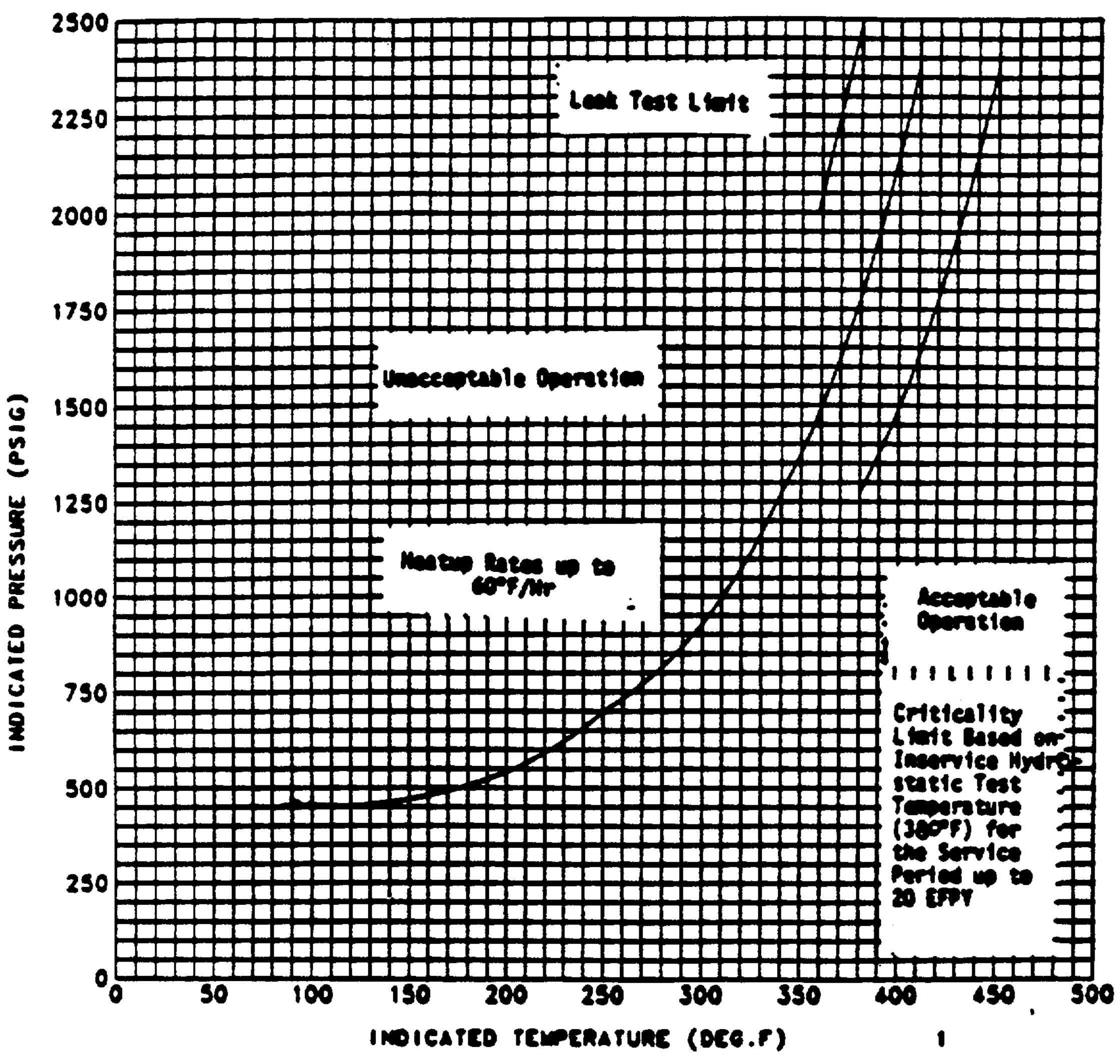
MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD[1]

INITIAL RT_{NDT}: 10°F[1]

RT_{NDT} AFTER 20 EFPY: 1/4T, 252.5°F
3/4T, 200.4°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 60°F/HR FOR THE SERVICE PERIOD UP TO 20 EFPY. NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.



Reactor Coolant System Heatup Limitations (60°F/Hr)
Applicable for the First 20 EFPY

Figure 3.1-1a

MATERIAL PROPERTY BASIS

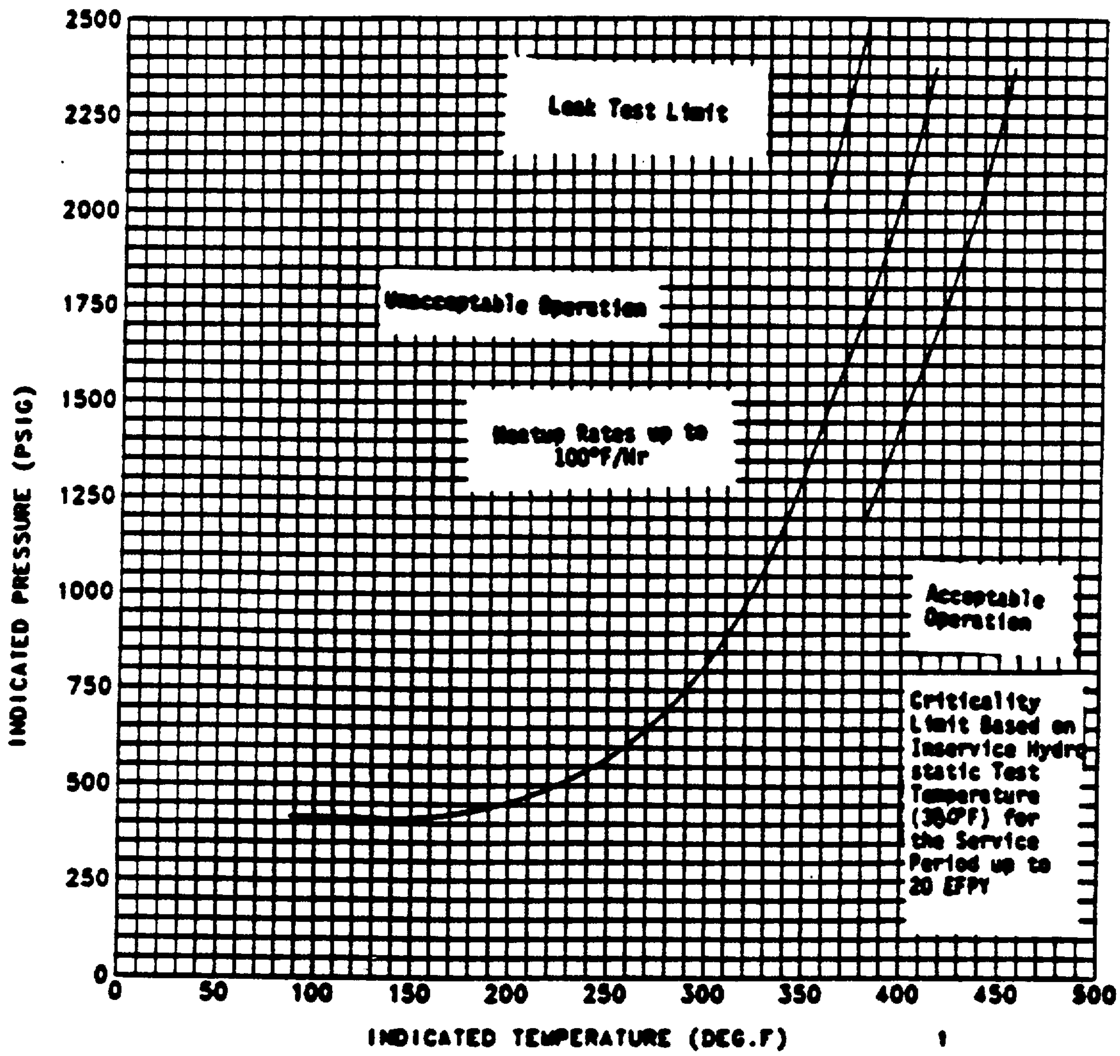
CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD[1]

INITIAL RT_{NDT}: 10°F[1]

RT_{NDT} AFTER 20 EFPY: 1/4T, 252.5°F

3/4T, 200.4°F

CURVES APPLICABLE FOR HEATUP RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 20 EFPY. NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.



Reactor Coolant System Heatup Limitations (100°F/Hr)
Applicable for the First 20 EFPY

Figure 3.1-1b

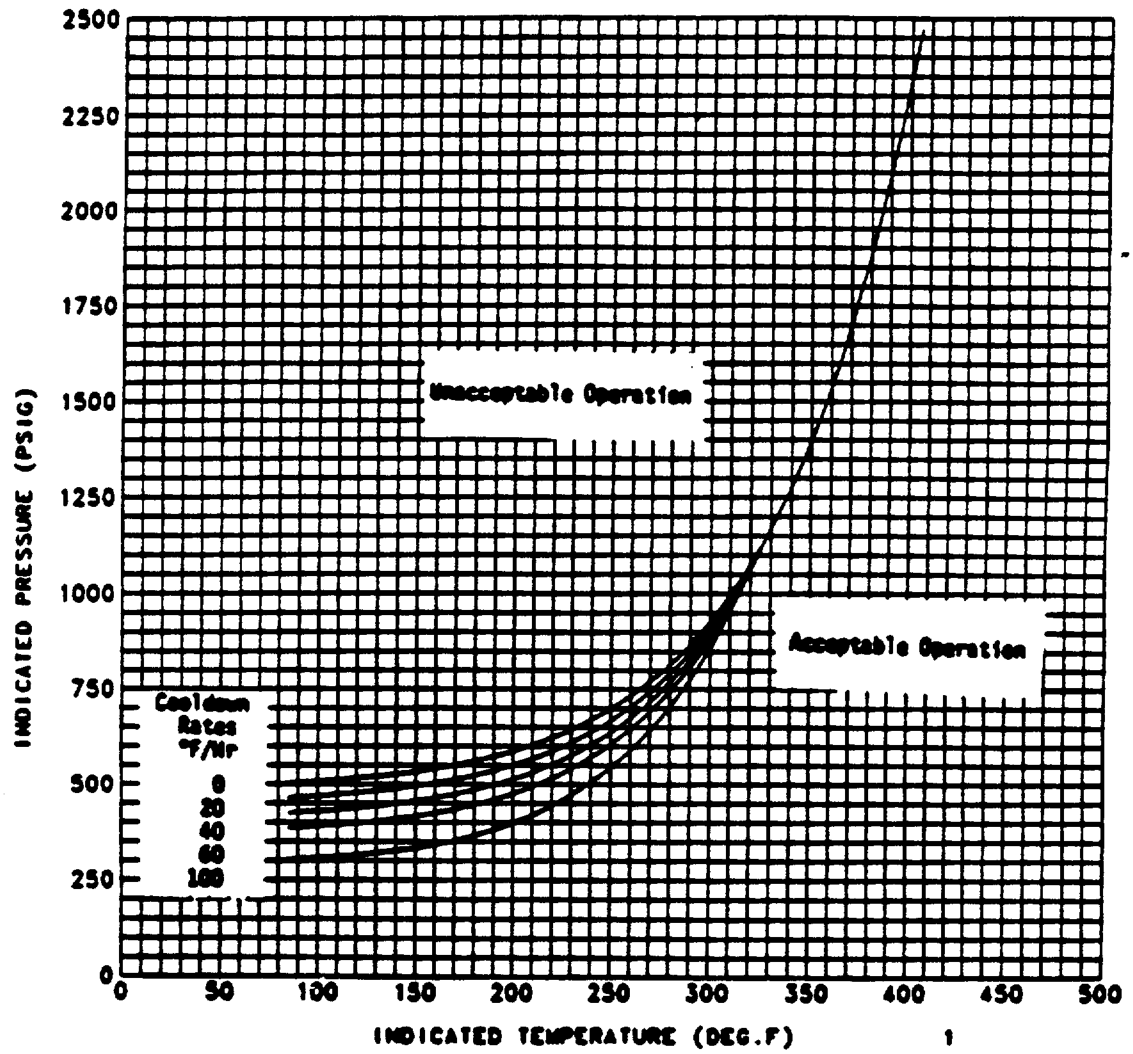
MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD[1]

INITIAL RT_{NDT}: 10°F

RT_{NDT} AFTER 20 EFPY: 1/4T, 252.5°F
3/4T, 200.4°F

CURVES APPLICABLE FOR COOLDOWN RATES UP TO 100°F/HR FOR THE SERVICE PERIOD UP TO 20 EFPY. NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.



Reactor Coolant System Cooldown Limitations Applicable for the First 20 EFPY

Figure 3.1-1c

2. BASES - PRESSURE/TEMPERATURE LIMITS

All components in the Reactor Coolant System (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 fracture toughness properties of the ferritic material in the reactor vessel were determined in accordance with the NRC Standard Review Plan, ASTM E185-73 and in accordance with additional reactor vessel requirements.

The properties are then evaluated in accordance with Appendix G of the 1983 Edition of Section III of the ASME Boiler and Pressure Vessel Code and the additional requirements of 10CFR50, Appendix G and the calculation methods described in Westinghouse Report GTSD-A-1.12, "Procedure for Developing Heatup and Cooldown Curves".

The heatup and cooldown limit curves, Figures 3.1-1a, 3.1-1b, and 3.1-1c 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 (20 EFPY).

The reactor vessel materials have been tested to determine their initial RT_{NDT}; the results of these tests as well as other material properties are shown in Tables B3.1-1 and B3.1-2. 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 limiting Reactor Vessel material has been predicted using Regulatory Guide 1.99, Revision 2, dated May 1988 (the latest accepted NRC methodology), "Radiation Embrittlement of Reactor Vessel Materials". The heatup and cooldown limit curves of Figures 3.1-1a, 3.1-1b, and 3.1-1c include predicted adjustments for this shift in RT_{NDT} at the end of the applicable service period.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10CFR Appendix H, reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. The surveillance specimen withdrawal schedule is shown in Specification 4.20.1. 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 material (Intermediate to Lower Shell Circumferential Weld) in Unit 3 and 4 is identical, the RV surveillance program was integrated and the results from capsule testing is applied to both Units. The surveillance capsule "T" results from Unit 3 (WCAP 8631) and Unit 4 (SWRI 02-4221) and the capsule "V" results from Unit 3 (SWRI 06-8576) were used with the methodology in Regulatory Guide 1.99 Revision 2 to provide limiting material properties information for generating the heatup and cooldown curves in Figures 3.1-1a, 3.1-1b, and 3.1-1c. 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.

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.

TABLE B3.1-1
REACTOR VESSEL TOUGHNESS DATA
TURKEY POINT - UNIT 3

Component	Material Type	Cu (%)	Ni (%)	P (%)	NDTT (°F)	50 ft lb/35 mils Lateral Expansion Temp (°F)		RT_{NDT} (°F)[a]	Minimum Upper Shelf (ft lb)	
						Long	Trans		Long	Trans
Cl. Hd. Dome	A302 Gr. B	-	-	0.010	0	-	36[a]	0	> 70	> 45.5[a]
Cl. Hd. Flange	A508 Cl. 2	-	0.72	0.010	44[a]	-	31[a]	44	>118	> 76.5[a]
Ves. Sh. Flange	A508 Cl. 2	-	0.65	0.010	-23[a]	-	-41[a]	-23	>120	> 78[a]
Inlet Nozzle	A508 Cl. 2	-	0.76	0.019	60[a]	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	0.74	0.019	60[a]	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	0.80	0.019	60[a]	-	NA	60	NA	NA
Outlet Nozzle	A508 Cl. 2	-	0.79	0.010	27[a]	-	9[a]	27	>110	> 71.5[a]
Outlet Nozzle	A508 Cl. 2	-	0.72	0.010	7[a]	-	-22[a]	7	>111	> 72[a]
Outlet Nozzle	A508 Cl. 2	-	0.72	0.010	42[a]	-	23[a]	42	>140	> 91[a]
Upper Shell	A508 Cl. 2	-	0.68	0.010	50	-	44[a]	50	>129	> 83.5[a]
Inter. Shell	A508 Cl. 2	0.058	0.70	0.010	40	-	25[a]	40	>122	> 79[a]
Lower Shell	A508 Cl. 2	0.079	0.67	0.010	30	-	2[a]	30	163	106[a]
Trans. Ring	A508 Cl. 2	-	0.69	0.013	60[a]	-	58[a]	60	>109	> 70.5[a]
Bot. Hd. Dome	A302 Gr. B	-	-	0.010	-10	-	NA	30	NA	NA
Inter. to Lower Shell Girth Weld	SAW	0.26	0.60	0.011	10[b]	-	63	10[b]	-	63
HAZ	HAZ	-	-	-	0[a]	-	0	0	-	168

(a) Estimated Values Based on NUREG-0800, Branch Technical Position - MTEB 52
(b) Actual Value

TABLE B 3.1-2
REACTOR VESSEL TOUGHNESS DATA
TURKEY POINT - UNIT 4

Component	Material Type	Cu (%)	Ni (%)	P (%)	NDTT (°F)	50 ft lb/35 mils Lateral Expansion Temp (°F) Long	Trans	RT_{NDT} (°F)[a]	Minimum Upper Shelf (ft lb) Long	Trans
Cl. Hd. Dome	A302 Gr. B	-	-	0.008	-20	-	NA	30	NA	NA
Cl. Hd. Flange	A508 Cl. 2	-	0.72	0.010	- 4[a]	-	27[a]	- 4	199	129[a]
Ves. Sh. Flange	A508 Cl. 2	-	0.68	0.010	- 1[a]	-	-11[a]	- 1	176	114[a]
Inlet Nozzle	A508 Cl. 2	0.08	0.71	0.009	60[a]	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	0.84	0.019	60[a]	-	NA	60	NA	NA
Inlet Nozzle	A508 Cl. 2	-	0.75	0.008	16[a]	-	13[a]	16	162	105[a]
Outlet Nozzle	A508 Cl. 2	-	0.78	0.010	7[a]	-	-25[a]	7	165	107[a]
Outlet Nozzle	A508 Cl. 2	-	0.68	0.010	38[a]	-	16[a]	38	160	104[a]
Outlet Nozzle	A508 Cl. 2	-	0.70	0.010	60[a]	-	42[a]	60	143	93[a]
Upper Shell	A508 Cl. 2	-	0.70	0.010	40	-	32[a]	40	156	101[a]
Inter. Shell	A508 Cl. 2	0.054	0.69	0.010	50	-	90[a]	50	143	93[a]
Lower Shell	A508 Cl. 2	0.056	0.74	0.010	40	-	38[a]	40	147	97[a]
Trans. Ring	A508 Cl. 2	-	0.69	0.011	60[a]	-	30[a]	60	NA	NA
Bot. Hd. Dome	A302 Gr. B	-	-	0.010	10	-	30[a]	10	NA	NA
Inter. to Lower Shell Girth Weld	SAW	0.26	0.60	0.011	10[b]	-	63	10[b]	NA	63
HAZ	HAZ	-	-	-	0		NA	0	NA	140

(a) Estimated Values Based on NUREG-0800, Branch Technical Position - MTEB 52
(b) Actual Value

B3.1.2.1 MODERATOR TEMPERATURE COEFFICIENT

The limitations on moderator temperature coefficient (MTC) are provided to ensure that the value of this coefficient remains within the limiting condition assumed in the FSAR accident and transient analyses.

The MTC values of this specification are applicable to a specific set of plant conditions; accordingly, verification of MTC values at conditions other than those explicitly stated will require extrapolation to those conditions in order to permit an accurate comparison.

The most negative MTC, value to the most positive moderator density coefficient (MDC), was obtained by incrementally correcting the MDC used in the FSAR analyses to nominal operating conditions. These corrections involved subtracting the incremental change in the MDC associated with a core condition of all rods inserted (most positive MDC) to an all rods withdrawn condition and, a conversion for the rate of change of moderator density with temperture at RATED THERMAL POWER conditions.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 134 TO FACILITY OPERATING LICENSE NO. DPR-31

AND AMENDMENT NO. 128 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 September 21, 1988, Florida Power and Light Company (the licensee) requested approval to revise Sections 3.1.2 and B3.1.2 in the Turkey Point Units 3 and 4 (TP3 and TP4) Technical Specifications. The purpose of this request was to revise the existing pressure-temperature limits and extend the operation period of the limits up to 20 effective full power years (EFPY). The existing limits are applicable up to 10 EFPY and will soon expire. It is estimated that TP3 will reach 10 EFPY early in 1989 and TP4 will reach 10 EFPY in mid-1989. The proposed pressure-temperature limits will govern operation of both units for another 10 EFPY. The purpose of the limits is to provide permissible pressure and temperature for the following operations: heatup, cooldown, and leak tests.

The NRC regulations and staff guidance applicable to the evaluation of pressure-temperature (P/T) limits include the following: Appendix A (GDC-31), 10 CFR 50.60, Appendices G and H to 10 CFR Part 50; ASTM E-185 and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); Regulatory Guide 1.99, Revision 2; and Standard Review Plan (SRP) Section 5.3.2.

Appendix A to 10 CFR Part 50 describes General Design Criteria (GDC) for nuclear power plants. Specifically, GDC-31 requires that the reactor coolant pressure boundary (which includes the reactor vessel) be designed to assure that (1) the boundary behaves in a non-brittle manner, and (2) the probability of rapidly propagating fracture is minimized. In 10 CFR 50.60, acceptance criteria are addressed for fracture prevention measures for normal operation. All lightwater nuclear power reactors are required by 10 CFR 50.60 to meet the requirements of Appendices G and H. Appendices G and H describe specific requirements for the reactor vessel which must be met to assure that GDC-31 and 10 CFR 50.60 are satisfied.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications (TS) for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the TS. The P/T limits are among the limiting conditions of operation in the TS for nearly all, if not all, plants in the

U.S. Appendices G and H to 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance. These must be considered in setting P/T limits.

Appendix G to 10 CFR Part 50 specifies fracture toughness requirements for ferritic materials and requires the licensee to test ferritic materials in accordance with the ASME Code and, in particular, to test the beltline materials in the surveillance capsules in accordance with Appendix H to 10 CFR Part 50. Appendix H, in turn, refers to ASTM E-185. These tests define the condition of vessel embrittlement at the time of capsule withdrawal in terms of the increase in the reference temperature (RT_{NDT}). Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted RT_{NDT} and upper shelf energy. A method that is acceptable to the NRC staff is described in Regulatory Guide 1.99, Revision 2.

Appendix H to 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to ASTM E-185 which, in turn, requires that the capsules be installed in the vessel before startup and that they contain test specimens that were made from plate, weld, and heat-affected-zone materials of the reactor beltline. Appendix H also considers an integrated surveillance program for a set of reactors that have similar design and operating features. The staff approved the TP3 and TP4 integrated surveillance program by letter dated April 22, 1985.

2.0 EVALUATION

The licensee had removed capsules S, T, and V from TP3 and capsules S and T from TP4 and had submitted material analyses of these capsules.

In a letter dated October 30, 1987, the NRC staff recommended that the surveillance test results from all capsules withdrawn from TP3 and TP4 be integrated in order to evaluate the effect of neutron irradiation on beltline materials of both vessels. This was consistent with use of the integrated surveillance program approved by the staff in 1985. A part of the Safety Evaluation contained in the October 30, 1987 letter indicates that the test results from the most limiting materials irradiated in capsules in TP3 and TP4 reactors can be used to determine the vessels' fracture toughness. Therefore, for this review the staff used data from previously removed capsules to calculate the adjusted RT_{NDT} in order to verify the licensee's adjusted RT_{NDT}. In calculating P/T limits, the limiting material is considered to be weld metal in the highest neutron fluence area (beltline) of the reactor vessel and which has the highest RT_{NDT}. The limiting beltline material for both TP3 and TP4 is the intermediate-shell-to-lower-shell girth weld SA-1101. This weld was done by a submerged arc welding process and the wire heat number was 71249 and the flux was Linde 80, Lot 8445. The limiting weld wire materials were used to make welds from which tensile and Charpy impact test specimens were prepared. These test specimens were encased in capsules T and V in TP3 and capsule T in TP4; therefore, the neutron fluence and measured increase in RT_{NDT} obtained from these capsules are valid for use in the staff's calculation (Table 1). The following surveillance data were reported by the licensee in the submittal dated September 21, 1988.

The copper and nickel contents of the limiting weld wire were estimated to be 0.26% and 0.60%, respectively; and the initial RT_{NDT} was measured to be 10°F. At the vessel inside radius, the neutron fluence for 20 EFPY was estimated to be 2.022 X 10^{19} n/cm². At the end of life, neutron fluence was estimated to be 2.79 X 10^{19} n/cm² for TP3 and 2.695 X 10^{19} n/cm² for TP4. The staff used the surveillance data to calculate the adjusted RT_{NDT} using the method in Section 2.1 of Regulatory Guide 1.99, Rev. 2. The adjusted RT_{NDT} is the sum of the initial RT_{NDT}, increase in RT_{NDT}, and margin of the limiting weld wire at the ¼T (T is the vessel thickness) limiting location. The staff's calculated adjusted RT_{NDT} agrees with the licensee's calculation as shown in Table 2.

In addition to an evaluation of limiting beltline materials, Appendix G also requires an evaluation of materials in the closure flange region. Section IV.2 of Appendix G states that when pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions that are highly stressed by the bolt preload must exceed the RT_{NDT} of the material in those regions by at least 120°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 44°F reported in the licensee's amendment proposal, dated September 21, 1988, for both TP3 and TP4, the staff has determined that the closure flange limits in the proposed pressure-temperature curves satisfy Appendix G to 10 CFR Part 50.

The cover letter of the licensee's application indicated that low temperature overpressure protection by the cold overpressure mitigation system (COMS) remained acceptable, and the PORV setpoint would remain at 415 psi under these conditions with the new curves. The NRC staff reviewed the curves and, combined with reference to an earlier Safety Evaluation dated December 23, 1982, the staff is satisfied that the P/T curves had not changed significantly for the heatup rate (0°F/hr.) and temperature (100°F) used as the bases for the PORV setpoint to provide low temperature overpressure protection.

The licensee proposed to reformat the existing requirements in TS 3.1.2 to explicitly state the limiting conditions for operation, applicability and action requirements, in order to be consistent with NUREG-0452, Standard Technical Specifications for Westinghouse Pressurized Water Reactors. In addition, the licensee proposed a change to the related pages in the "Bases" section of the TS to provide additional understanding and to make it consistent with the rest of the amendment proposal. The NRC staff finds the proposed format and the revised bases are acceptable because they are administrative changes which have little safety significance.

The staff has concluded that the proposed pressure-temperature limits on the reactor coolant system for heatup, cooldown, and leak tests are in conformance with requirements of Appendix G to 10 CFR Part 50. The proposed limits are acceptable up to 20 EFPY because, based on the staff's evaluation, fracture toughness of the reactor vessels required for setting P/T limits is in accordance with Appendix G to 10 CFR Part 50 and the other applicable requlations stated above. The limits may be incorporated into the Turkey Point Units 3 and 4 Technical Specifications.

3.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION

The licensee's request for amendments to the operating licenses for the Turkey Point Plant, Unit Nos. 3 and 4, including a proposed determination by the staff of no significant hazards consideration, was noticed in the Federal Register on October 19, 1988. Because the staff received a request for hearing on this issue, the comments of the intervenor were considered in making a final no significant hazards determination. This is the staff's final determination of no significant hazards consideration.

The Commission's regulations in 10 CFR 50.92(c) include three standards used by the NRC staff to arrive at a determination that a request for amendment involves no significant hazards considerations. These regulations state that the Commission may make such a final determination if operation of a facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

In its submittal, the licensee has evaluated the proposed change in accordance with the standards of 10 CFR 50.92(c) and has determined that operation of Turkey Point Units 3 and 4 in accordance with the proposed amendments would not:

"(1) Involve a significant increase in the probability or consequences of an accident previously evaluated.

The pressure/temperature (P/T) limit curves in the Technical Specifications are conservatively generated in accordance with the fracture toughness requirements of 10 CFR [Part] 50, Appendix G as supplemented by Appendix G of Section III of the ASME Boiler and Pressure Vessel Code. The RT_{NDT} values for the revised curves are based on Regulatory Guide 1.99, Revision 2, dated May 1988, as discussed in Westinghouse Electric Corporation Report titled "Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation." The analysis of reactor vessel material irradiation surveillance specimen revised curves in conjunction with the surveillance specimen program ensures that the reactor coolant pressure boundary will behave in a non-brittle manner and that the possibility of rapidly propagating fracture is minimized.

"The revised pressure/temperature limit curves do not represent a significant change in the configuration or operation of the plant and thus do not involve an increase in either the probability or the consequences of accidents previously evaluated.

"(2) Create the possibility of a new or different kind of accident.

The analysis performed has resulted in revised P/T limits based on the fracture toughness requirements of 10 CFR [Part] 50, Appendix G. Since there is no significant change in the configuration or operation of the facility due to the proposed amendment, use of the revised P/T limits will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"(3) Involve a significant reduction in a margin of safety.

The proposed change will not involve a significant reduction in a margin of safety because the requirements of 10 CFR [Part] 50, Appendix G are satisfied.

"In addition, with respect to the reformatting change, the Commission has provided guidance concerning the application of the standards for determining whether a significant hazards consideration exists by providing certain examples 51 FR 7751 of amendments that are considered not likely to involve a significant hazards consideration. Example (i) relates to a purely administrative change to Technical Specifications: for example, a change to achieve consistency throughout the Technical Specifications, correction of an error, or a change in nomenclature. The proposed change reformatting the existing requirements in TS 3.1.2 is similar to example (i) in that it is an administrative change which states the requirements in a format consistent with that of the Standard Technical Specifications and does not involve technical or plant modifications.

"Therefore, operation of the facility in accordance with the proposed amendment would pose no threat to the public health and safety, and would not involve a significant hazards consideration."

The NRC staff performed its own evaluation (below) of no significant hazards consideration in accordance with 10 CFR 50.92, taking into account the licensee's evaluation above as well as public comments received. The issue for consideration in setting P/T limits is loss of reactor vessel integrity due to brittle fracture. Revising the P/T limits would not change any consequences of a failed reactor vessel. The P/T limits only bear on whether vessel integrity is lost. Therefore, this portion of the first standard is satisfied. Also, because the proposed amendments merely revise existing P/T limits, no new or different kind of accident would be involved. The issue for consideration remains a brittle fracture-induced loss of vessel integrity. Thus, the second standard is satisfied. This reduces the no significant hazards consideration to a determination of whether there is a significant increase in probability of loss of vessel integrity, or whether there is a significant reduction in a margin of safety. Safety margins are maintained for reactor vessel integrity and are not reduced by revising P/T limits. For example, in calculating the actual stress to which a reactor vessel is subjected, the staff assumes the pressure stress component to be doubled for heatup and cooldown, and assumes a crack to be present which extends 1/4 of the distance through the vessel wall thickness. As another example, in estimating the fracture toughness for the vessel, the conservative lower bound curve (as presented in the ASME Code) is used, rather than the mean value. These assumptions are examples of safety margins which are standard requirements of the staff and do not undergo a reduction because of revised P/T limits. Because the margins of safety are maintained in revising the P/T limits, the probability of loss of vessel integrity is not significantly changed. Thus, the proposed amendment does not involve a significant increase in the probability of an accident previously evaluated or a significant reduction in a margin of safety.

Further discussion is provided below which explains aspects of the licensee's surveillance programs and shows that the licensee meets the relevant staff requirements.

The fracture toughness of the steel in a reactor pressure vessel wall is determined primarily by the following factors: (1) the particular material (composition and metallurgical history), (2) the accumulated irradiation level (neutron fluence) to which the material is exposed, and (3) the temperature of the material. In a reactor pressure vessel, significant loadings result from the internal pressure and thermal gradient through the vessel wall thickness during heatup and cooldown. Since the fracture toughness of the vessel material decreases with decreasing temperature, P/T limits are required during normal reactor operation and tests to control operational stresses to the reactor vessel. Furthermore, because the fracture toughness of the vessel material decreases with increasing neutron irradiation (i.e., time duration of operation), a material surveillance program is required to monitor changes in the fracture toughness properties of the reactor vessel beltline material over the lifetime of the vessel. The P/T limits are periodically revised to take into account additional test data from the surveillance program on the changes in the fracture toughness properties due to irradiation.

Neutron embrittlement for Turkey Point Units 3 and 4 is being monitored through an integrated surveillance program, which is in compliance with Appendix H to 10 CFR Part 50 and was approved by the staff in a letter dated April 22, 1985. The benefits of the integrated program include having more capsules available that are applicable to each reactor unit. In each vessel, there are capsules containing the critical weld material. Under the integrated surveillance program, the test results from all capsules will be applied to vessel integrity analyses for both units. The twin units 3 and 4 at Turkey Point are nearly identical in their design, construction, reactor vessel materials, operating procedures and neutron flux spectra. The integrated surveillance program provides the best use of the available surveillance capsules containing the critical weld material for both units. The surveillance program for Turkey Point Units 3 and 4 comprises a set of capsules in each reactor vessel containing samples of the weld materials and base metals used in fabricating the beltline of the reactor vessel. Thus, the surveillance samples removed to obtain test data have the same composition and metallurgical history as the materials in the reactor vessel wall. The critical (most embrittled) material for Turkey Point Units 3 and 4 reactor vessels are the center girth welds which are positioned at about midheight of the reactor cores (the region of the highest neutron fluence). Fabrication records show that the center girth weld for Unit 3 and the center girth weld for Unit 4 were made with the same materials, that is, the same weld wire heat and the same weld flux lot. The surveillance welds made for Unit 3 test specimens were made with the same materials as the center girth welds in both reactor vessels. The surveillance welds for Unit 4 test specimens were made with weld wire from the same heat of material but different flux lot than the center girth welds in both reactor vessels. Although the Unit 4 surveillance weld specimens were fabricated using a different flux lot, the weld specimens were considered to be representative of the girth welds in both reactor vessels because flux lot number is only of minor importance in determining the sensitivity to irradiation embrittlement. Based on the similarity between materials in the center girth welds and the materials used to fabricate the surveillance weld specimens, the test results from capsules in either Unit 3 or 4 can be used to monitor the neutron embrittlement in both reactor vessels.

The staff recommends that the licensee estimate neutron irradiation embrittlement by the method contained in Regulatory Guide 1.99, Revision 2, dated May 1988. Regulatory Guide 1.99, Revision 2 contains equations and margins for safety which account for uncertainties in calculating neutron embrittlement and presents a method for evaluating the surveillance test data. The results of Charpy tests from many reactors were compiled to form a surveillance data base. This data base was used by the staff to develop an equation to calculate individual vessel embrittlement. For example, the effects of irradiation and nil-ductility temperature are included in the calculation of P/T limits to insure conservatism. These methods assure that the calculated effects of irradiation nil-ductility temperature provide a conservative basis for determining P/T limits, and do not underestimate the irradiation damage to the reactor vessel welds.

The licensee used the equations set forth in Regulatory Guide 1.99, Revision 2 to calculate the embrittlement of welds in the reactor vessels at the Turkey Point Units 3 and 4. The weld material surveillance data from both units, obtained through the integrated surveillance program, were used in calculating embrittlement projections. The data from Capsules T and V in Unit 3 were obtained at a Charpy energy level of 30 ft-lb and the data from Capsule T in Unit 4 were adjusted to a Charpy energy level of 30 ft-lb from a 42 ft-lb level. The weld material sample in Unit 4, Capsule T, showed a degree of embrittlement which is greater than the mean embrittlement projected for the weld material.

The greater than expected embrittlement for one weld material sample from Unit 4 does not demonstrate that the beltline material in Unit 4 is as embrittled as that sample. The Unit 4 data point is within the uncertainty and scatter that can be expected from measurements of this type. The issue of how to properly analyze the test results from the sample in Capsule T, Unit 4 was addressed by Dr. Pryor N. Randall in an affidavit prepared for an earlier Turkey Point proceeding.* As part of his discussion, Dr. Randall points out the Commission regulations require that measurements of capsule test samples be taken at a Charpy energy level of 30 ft-lb. The 30 ft-lb level more accurately reflects the degree of vessel embrittlement. For these reasons, the NRC staff adjusted the 1976 test results from Capsule T in Unit 4 to a Charpy energy level of 30 ft-lb from a 42 ft-lb level. This adjusted value was closer to the value obtained at 30 ft-lbs for capsules T and V in Unit 3. Dr. Randall states that a more accurate measure of embrittlement of the critical weld in Unit 4 is obtained by using the Unit 3 sample, corrected for differences from Unit 4, than by using the Unit 4 samples of different weld material. This is because no samples of the critical weld lot for Unit 4 were put in a Unit 4 capsule, although some were put in the capsules in Unit 3.

Based on the projected degree of neutron embrittlement at the end of 20 effective full power years (EFPY), which was estimated using Regulatory Guide 1.99, Revision 2, the licensee submitted P/T limits for Turkey Point Units 3 and 4 for application up to 20 EFPY. The Turkey Point plants are currently at about 10 EFPY.

---

*Declaration of Pryor N. Randall, dated December 2, 1985, filed with U.S. Court of Appeals for the District of Columbia Circuit in Lorion vs. U.S. Nuclear Regulatory Commission, No. 82-1132.

The licensee is conservative by committing to operate between now and the end of 20 EFPY using P/T limits based on the embrittlement level at the end of 20 EFPY. The NRC staff performed independent P/T limit calculations according to guidance in NRC Standard Review Plan 5.3.2, containing staff required margins. The staff's calculations determined that the licensee's submittal was acceptable and that the neutron embrittlement calculation was in accordance with Regulatory Guide 1.99, Revision 2.

In a preliminary assessment of the PTS issue, as shown in Table P.1 in NRC SECY-82-465, dated November 23, 1982, Turkey Point Units 3 and 4 were listed as having the third and second highest PTS screening nil-ductility temperature for all plants, respectively. However, this is no longer the case. The result of this preliminary assessment is no longer applicable because of updated chemistry and fluence data provided to, and accepted by, the NRC staff. Since then, the staff has established a PTS screening criterion in 10 CFR 50.61. The staff used the results of Charpy tests from many plants to develop an equation that can be used to calculate individual vessel embrittlement. The staff reviewed and accepted the January 23, 1986 submittal by the licensee for the Turkey Point Units 3 and 4 PTS evaluations based on additional data on the critical material composition and neutron fluence. The NRC staff summarized the results of the PTS findings in "Regulatory Analysis for Revision 2 to Regulatory Guide 1.99," dated November 20, 1987. For the Turkey Point Units 3 and 4 reactor vessels, the estimated value to be compared against the PTS screening criterion at the end of life (the present license expires in the year 2007) was 263°F, which is below the PTS screening criterion of 300°F for the Turkey Point vessels. The screening criterion of 300°F is prescribed by 10 CFR 50.61, based upon the limiting circumferential girth weld in the beltline region (there are no axial welds in this region). Turkey Point Units 3 and 4 will reach the PTS screening criterion in the year 2035. Because the NRC staff intends to amend the PTS screening criterion using newer Regulatory Guide 1.99, Revision 2 procedures, the staff also performed another PTS evaluation based on Regulatory Guide 1.99, Revision 2. For Turkey Point Units 3 and 4, the estimated value at the end of the present license in the year 2007 would be 283°F and would reach the PTS screening criterion of 300°F in the year 2020.

Based on the staff's evaluation as discussed above, the staff finds the revised P/T limits for Turkey Point Units 3 and 4 to be in compliance with Appendix G to 10 CFR Part 50. It is routine for licensees to revise their P/T limits based on the latest information available from reactor vessel materials surveillance programs. Furthermore, the integrated surveillance program at Turkey Point Units 3 and 4 complies with Appendix H to 10 CFR Part 50 and the surveillance test data have been evaluated in accordance with Regulatory Guide 1.99, Revision 2. In addition, the staff has found that the Turkey Point Units 3 and 4 reactor vessels' critical material will remain below the staff's PTS screening criterion for their licensed life which assures continued safe operation of both units, and therefore meets the requirements of 10 CFR 50.61. The earlier and the revised P/T limits comply with all of the applicable requirements above, and the revised P/T limits do not change the probability or consequences of an accident previously evaluated, do not create the possibility of a new or different kind of accident, and do not involve a reduction in a margin of safety.

Finally, for Turkey Point Units 3 and 4, the changes in format are administrative and the revised "Bases" reflects the revised P/T limits and provides a better understanding for the specific TS but does not impact the plant configuration or operation. Therefore these changes do not: (1) involve a significant increase in the probability or consequences of an accident previously evaluated, (2) create the possibility of a new or different kind of accident, or (3) involve a significant reduction in a margin of safety.

The staff has treated the statements made in the intervention petition of the Center for Nuclear Responsibility and Joette Lorion, dated November 17, 1988, as comments on the staff's proposed no significant hazards determination. The petition included seven numbered paragraphs. The first four numbered paragraphs simply identified the party intervening. The seventh numbered paragraph requested a hearing. Thus, these paragraphs do not bear on the no significant hazards consideration.

Paragraphs 5 and 6 were comments on the proposed no significant hazards determination and are discussed below. Comment numbers 5 and 6.a) merely stated that the three standards in 10 CFR 50.92(c) would be violated and therefore the proposed amendment involves a significant hazards consideration. Those comments are refuted by this Final No Significant Hazards Consideration Determination.

Comment 6.b) states that the use of data from the integrated surveillance program, and specifically the use of Unit 3 data to predict P/T limits for Unit 4, is scientifically invalid, not conservative, and increases the probability of an accident. The licensee met the requirements of Appendices A (GDC-31), G and H and followed the staff guidance in Regulatory Guide 1.99, Revision 2, in developing the P/T limits for Turkey Point Units 3 and 4. Therefore, the data used and the calculations derived from the data are scientifically valid, properly conservative and do not increase the probability of an accident. The NRC staff determined that the margins of safety have been maintained as required by the Commission's regulations.

Comment 6.c) again states that the proposed P/T limits for Units 3 and 4 are not conservative because of uncertainties in some estimates and calculations of the effects of irradiation and nil-ductility temperature for Unit 3, and therefore would increase the probability and consequences of an accident caused by pressurized thermal shock and pressure vessel rupture. As noted above, margins of safety in the regulations and staff guidance provide conservatism, and prevent a significant increase in probability of an accident, the licensee meets the staff's screening criteria in 10 CFR 50.61, and revising P/T limits does not change the consequences of a vessel rupture.

Comment 6.d) states that revised P/T limits will cause 10 CFR Part 50 Appendix G to be violated because of a significant reduction in a margin of safety. As noted above, revising the P/T limits as proposed is in accordance with regulations, specifically Appendix G, and safety margins are maintained.

Comment 6.e) states that Units 3 and 4 have the second and third most embrittled reactor vessel welds in the United States. As noted above, this is no longer the case. Also, a relative ranking of embrittlement among different reactors does not imply a safety problem. What is important is not whether a vessel is the most (or least) embrittled among others, but whether the degree of embrittlement is unacceptable. The Turkey Point Units 3 and 4 vessels have been shown to meet the NRC regulations and the staff's guidance governing brittle fracture and are acceptable. Comment 6.e) also states that these units are extremely close to the NRC screening criterion and it is unwise and not conservative to set P/T limits for a 10-year period. As discussed above, the staff's screening criteria are quite conservative and the Unit 3 and 4 vessels will not approach them until the year 2020. Furthermore, the NRC staff monitors changes in the nil-ductility temperature. For example, 10 CFR Part 50 Appendix H.III requires the licensee to report test results periodically to NRC. 10 CFR 50.61(b) requires the licensee to provide updated nil-ductility temperature projections. The P/T limits have been revised using conservative methods and the NRC staff, as well as the licensee, will monitor vessel embrittlement throughout the life of the plant, as is done with all other U.S. commercial nuclear power reactors.

For these reasons, and those given (above) by the licensee, the staff agrees with the licensee's determination, and therefore has made a final determination that the amendments do not involve a significant hazards consideration.

4.0 ENVIRONMENTAL CONSIDERATION

These amendments involve changes in the installation or use of the facilities components located within the restricted areas as defined in 10 CFR Part 20. The staff has determined that these 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 these amendments involve no significant hazards consideration. However, a request for hearing was received which included comments pertaining to no significant hazards consideration. Therefore, a final evaluation was made (above) of no significant hazards considerations, taking into account the comments received in the hearing request. 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 these amendments.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) these amendments will not (a) significantly increase the probability or consequences of an accident previously evaluated, (b) create the possibility of a new or different kind of accident from any accident previously evaluated, or (c) significantly reduce a margin of safety, and therefore, the amendments do not involve significant hazards consideration; (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Dated: January 10, 1989

Principal Contributors:

J. Tsao
G. E. Edison

Table 1 Surveillance Data

Capsules	Fluence, n/cm²	Increase in RT_{NDT}, °F
T, Unit 3	5.68×10^{18}	155
T, Unit 4	6.05×10^{18}	225
V, Unit 3	1.229×10^{19}	180

Table 2 The Adjusted RT_{NDT} for the Girth Weld at ¼T and 20 EFPY

Intermediate Shell to Lower Shell Girth Weld SA-1101 Heat Number 71249	Staff Calculation	Licensee Calculation
Copper, %	0.26	0.26
Nickel, %	0.60	0.60
Capsule fluence, n/cm²	2.022×10^{19}	2.022×10^{19}
Chemistry factor	200.2	200.2
Initial RT_{NDT}, °F	10	10
Increase in RT_{NDT}, °F	213	214.5
Margin, °F	28	28
Adjusted RT_{NDT}, °F	251	252.5

UNITED STATES NUCLEAR REGULATORY COMMISSION

FLORIDA POWER AND LIGHT COMPANY

DOCKET NOS. 50-250 AND 50-251

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES AND FINAL DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 134 and 128 to Facility Operating License Nos. DPR-31 and DPR-41, respectively, to the Florida Power and Light Company (the licensee), which revised the Technical Specifications for operation of the Turkey Point Plant, Unit Nos. 3 and 4, located in Dade County, Florida. The amendments were effective as of the date of issuance.

The amendments revise Section 3.1.2 of the Technical Specifications (TS) by incorporating revised pressure and temperature (P/T) limits for the Reactor Coolant System and pressurizer. The P/T limits currently in the TS are applicable up to 10 effective full power years, and will soon expire. The amendments replace these P/T curves with revised curves applicable up to 20 effective full power years. The amendments also revise the applicable "Bases" discussion to be consistent with the new limits, and reformat the TS to be consistent with more recent standards TS.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments.

Notice of Consideration of Issuance of Amendments and Proposed No Significant Hazards Consideration Determination and Opportunity for Hearing in connection with this action was published in the FEDERAL REGISTER on October 19, 1988 (53 FR 40988). A request for a hearing was filed on November 17, 1988 by Joette Lorion.

Under its regulations, the Commission may issue and make an amendment immediately effective, notwithstanding the pendency before it of a request for a hearing from any person, in advance of the holding and completion of any required hearing, where it has determined that no significant hazards considerations are involved.

The Commission has applied the standards of 10 CFR 50.92 and has made a final determination that the amendments involve no significant hazards considerations. The basis for this determination is contained in the Safety Evaluation related to this action. Accordingly, as described above, the amendments have been issued and made immediately effective and any hearing will be held after issuance.

The Commission has determined that the amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment was prepared for these amendments.

For further details with respect to the action, see (1) the application for amendments dated September 21, 1988, (2) Amendment Nos. 134 and 128 to Facility Operating License Nos. DPR-31 and DPR-41, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 2120 L Street, N.W., Washington, D.C., and at the Environmental and Urban Affairs Library, Florida International University, Miami, Florida 33199. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Projects I/II.

Dated at Rockville, Maryland this 10th day of January 1989.

FOR THE NUCLEAR REGULATORY COMMISSION

G E Edison

Gordon E. Edison, Sr. Project Manager
Project Directorate II-2
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation