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

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

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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} Δk/k/°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} Δk/k/°F from HZP to 70% RATED THERMAL POWER condition; and

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

d) Less negative than -3.5×10^{-4} Δk/k/°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.

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* 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

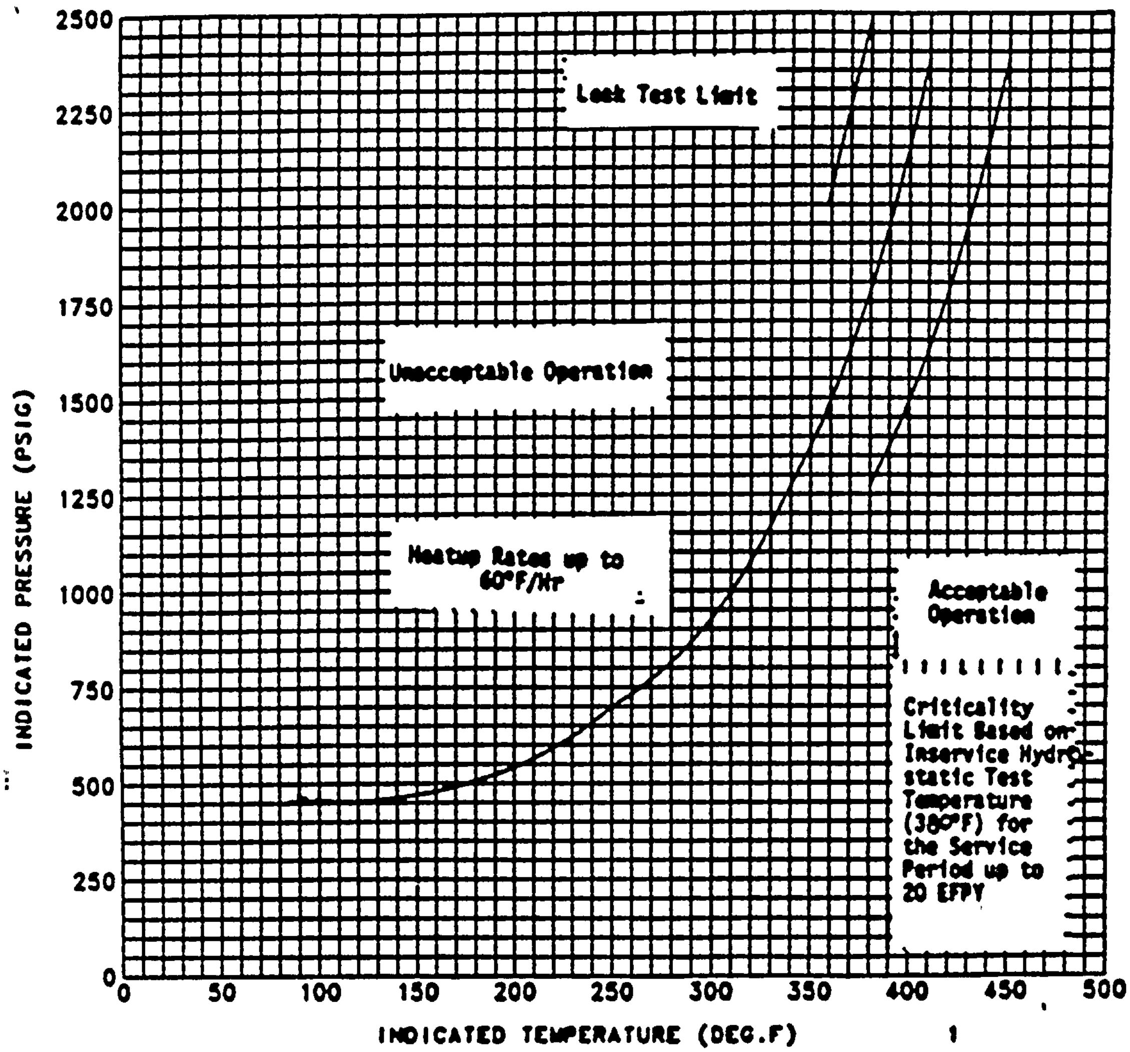
CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD[1]

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

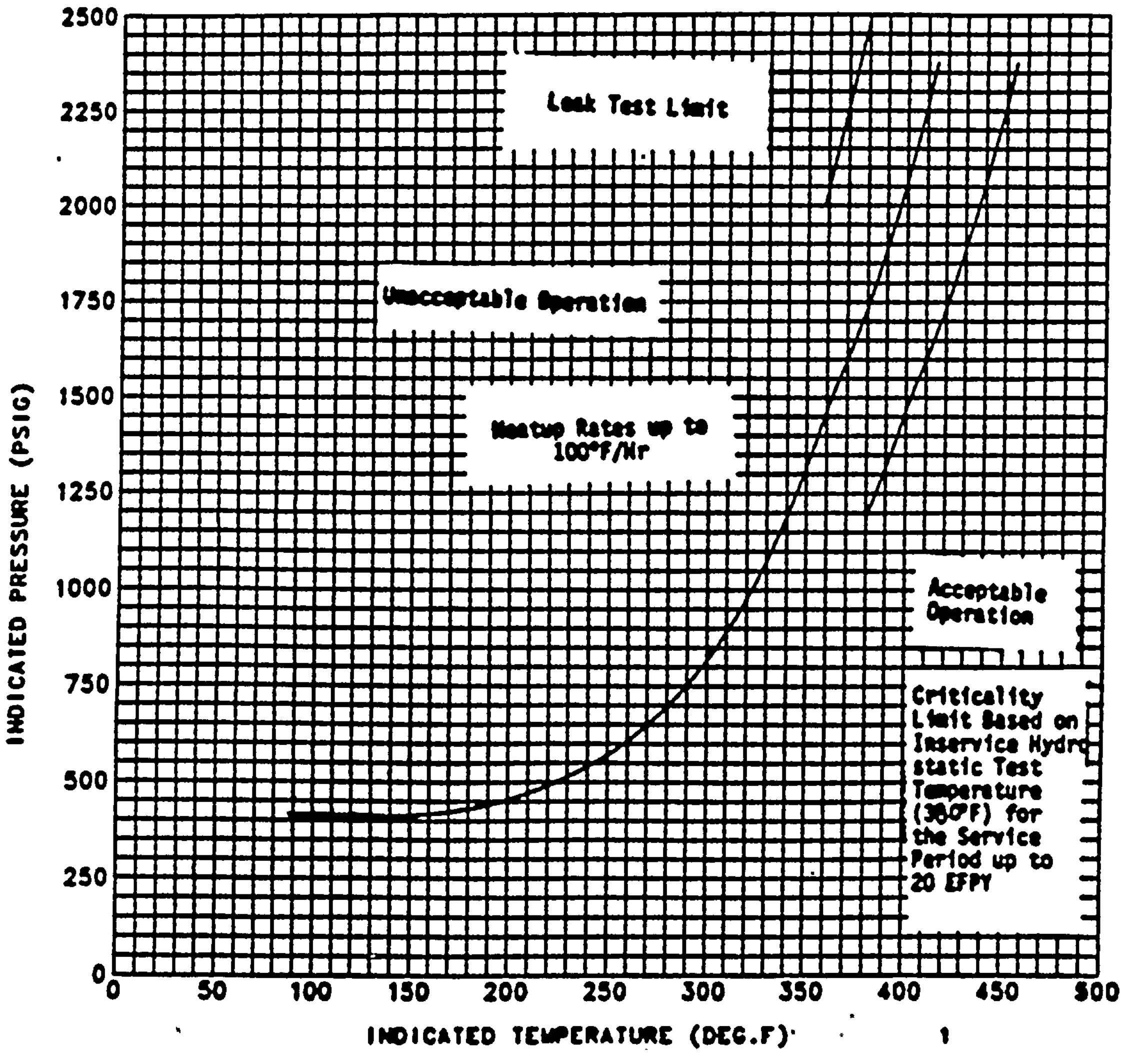
Figure 3.1-1a

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

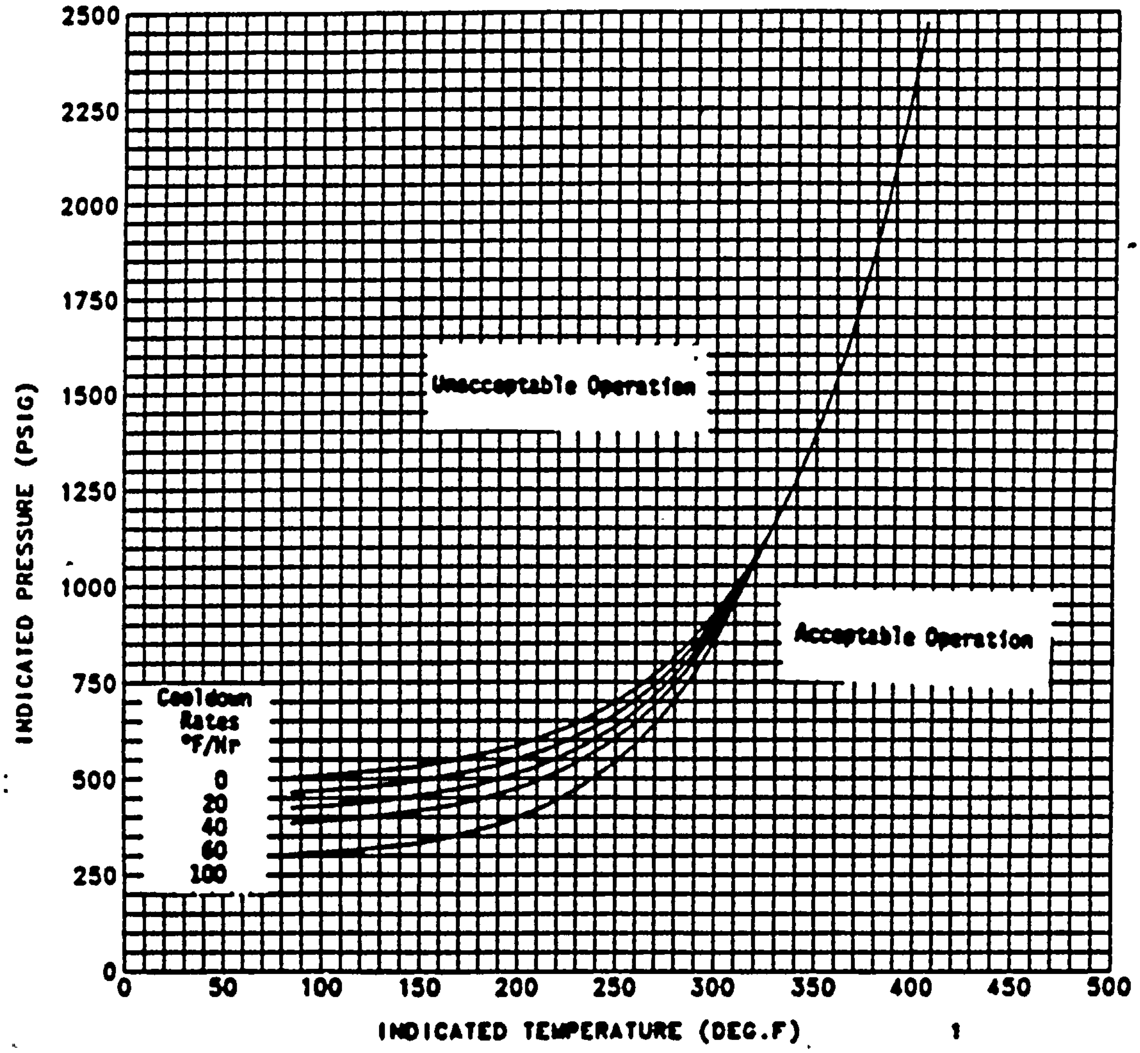
CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD[1]

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) Long	Trans	RT_{NDT} (°F)[a]	Minimum Upper Shelf (ft lb) 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	50 ft lb/35 mils Lateral Expansion Temp (°F) Trans	RT_{NDT} (°F) (a)	Minimum Upper Shelf (ft lb) Long	Minimum Upper Shelf (ft lb) 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.