<u>ATTACHMENT 3</u>

PROPOSED TECHNICAL SPECIFICATIONS

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License Conditions
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CORE OPERATING LIMITS REPORT

A. Pursuant to Section 104b of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location on the Turkey Point site;

B. Pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

C. Pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

D. Pursuant to the Act and 10 CFR Part 30 to receive, possess, and use at any time 100 millicuries each of any byproduct material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

E. Pursuant to the Act and 10 CFR Part 40 and 70 to receive, possess, and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactively contaminated apparatus;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units Nos. 3 and 4.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations; 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50, and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect, and is subject to the additional conditions specified below:

A. Maximum Power Level

reactor core

The applicant is authorized to operate the facility at ~~steady state~~ power levels not in excess of ~~2200~~ megawatts (thermal).

2300

C. Pursuant to the Act and 10 CFR Parts 30, 40, and 70 to receive, possess, and use at any time any byproduct, source and special nuclear material as sealed sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required:

D. Pursuant to the Act and 10 CFR Part 30 to receive, possess, and use at any time 100 millicuries each of any byproduct material without restriction to chemical or physical form, for sample analysis or instrument calibration;

E. Pursuant to the Act and 10 CFR Parts 40 and 70 to receive, possess and use at any time 100 milligrams each of any source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration;

F. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Turkey Point Units No. 3 and No. 4.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission Regulations in 10 CFR Part 20, Section 30.34 of 10 CFR Part 30, Section 40.41 of 10 CFR Part 40, Section 50.54 and 50.59 of 10 CFR Part 50 and Section 70.32 of 10 CFR Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

A. Maximum Power Level

The reactor shall not be made critical until the tests described in the applicant's letter of April 3, 1973, have been satisfactorily completed. Thereafter, the applicant is authorized to operate the facility at reactor core power levels not in excess of ~~2200~~ megawatts thermal.

2300

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 169 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.

4300

QUADRANT POWER TILT RATIO

1.23 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

RATED THERMAL POWER

1.24 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 2200 MWt.

2300

REPORTABLE EVENT

1.25 A REPORTABLE EVENT shall be any of those conditions specified in Section 50.73 of 10 CFR Part 50.

SHUTDOWN MARGIN

1.26 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

SITE BOUNDARY

1.27 The SITE BOUNDARY shall mean that line beyond which the land or property is not owned, leased, or otherwise controlled by the licensee.

SOLIDIFICATION

1.28 SOLIDIFICATION shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements.

SOURCE CHECK

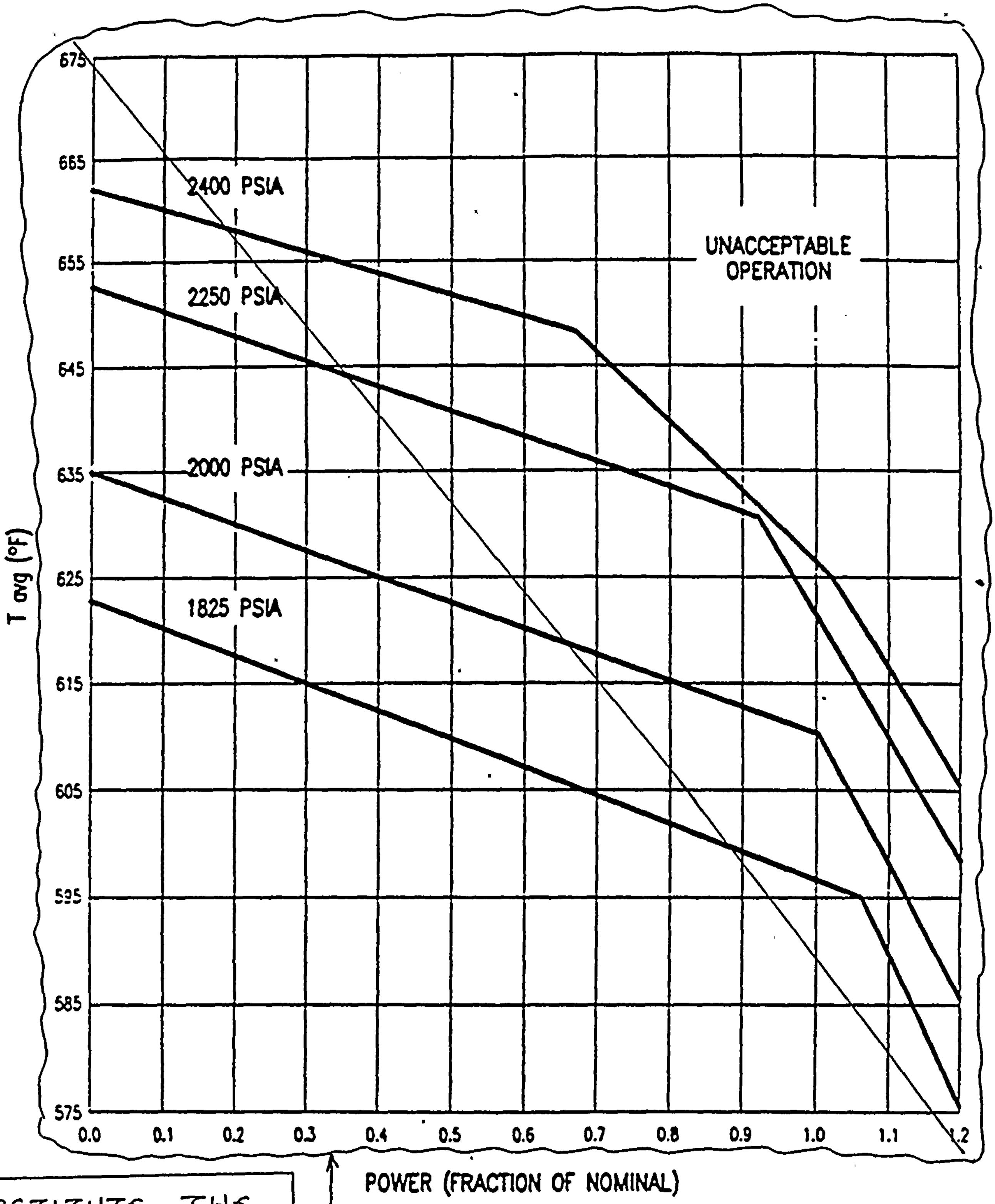
1.29 A SOURCE CHECK shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

STAGGERED TEST BASIS

1.30 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains, or other designated components obtained by dividing the specified test interval into n equal subintervals, and

b. The testing of one system, subsystem, train, or other designated component at the beginning of each subinterval.



SUBSTITUTE THE ATTACHED FIGURE FOR THIS FIGURE

FIGURE 2.1-1

REACTOR CORE SAFETY LIMIT - THREE LOOPS IN OPERATION

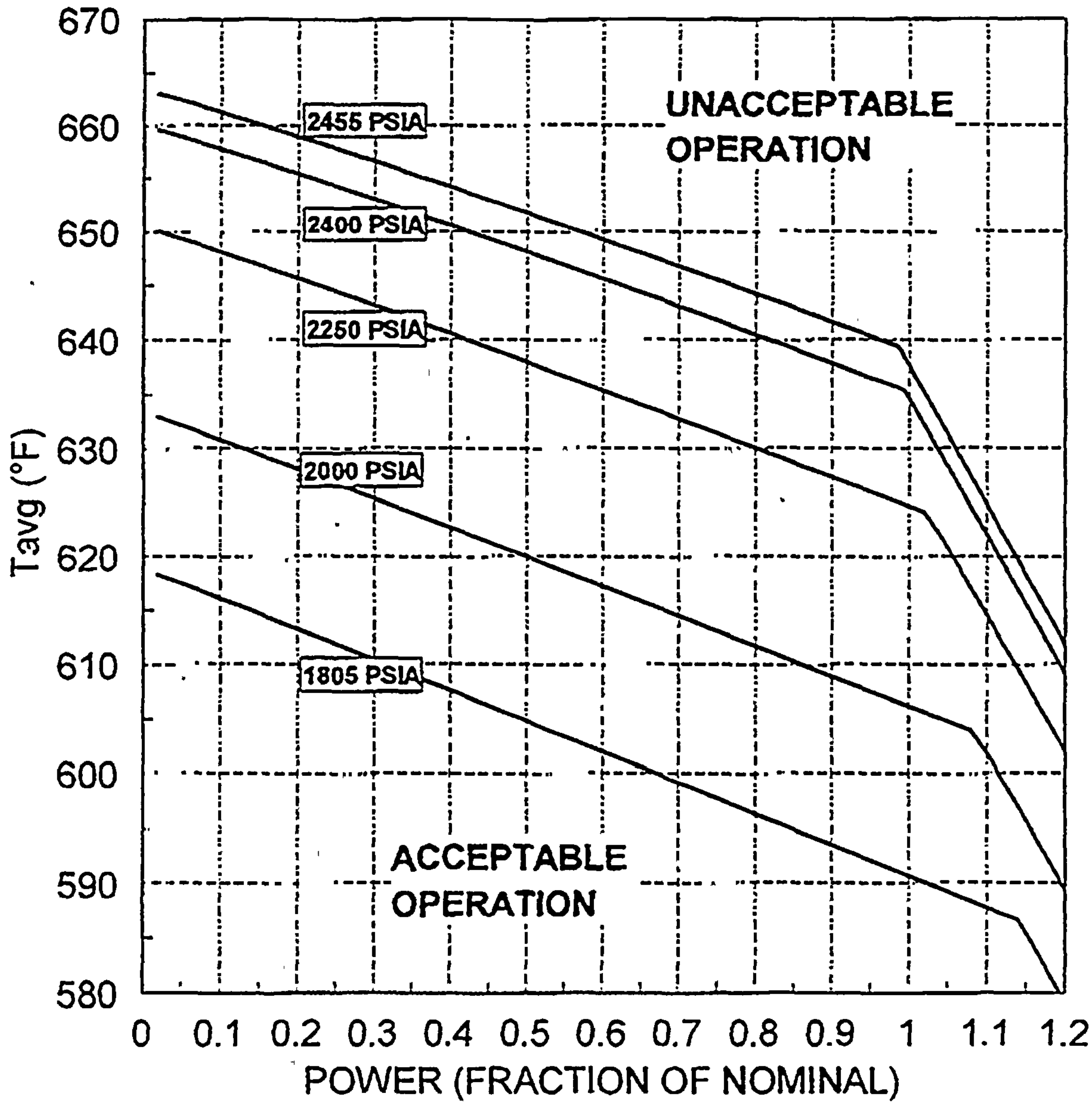


Figure 2.1-1
Reactor Core Safety Limit -Three Loops in Operation

TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
1. Manual Reactor Trip	N.A	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	≤112.0% of RTP**	≤109% of RTP**
b. Low Setpoint	≤28.0% of RTP**	≤25% of RTP**
3. Intermediate Range, Neutron Flux	≤31.0% of RTP**	≤25% of RTP**
4. Source Range, Neutron Flux	≤1.4 X 10^5 cps	≤10^5 cps
5. Overtemperature ΔT	See Note 2	See Note 1
6. Overpower ΔT	See Note 4	See Note 3
7. Pressurizer Pressure-Low	≥1817 psig	≥1835 psig
8. Pressurizer Pressure-High	≤2403 psig	≤2385 psig
9. Pressurizer Water Level-High	≤92.2% of instrument span	≤92% of instrument span
10. Reactor Coolant Flow-Low	≥~~88.7%~~ 88.8 %* of loop design flow*	≥90% of loop design flow*
11. Steam Generator Water Level Low-Low	≥~~13.2%~~ ~~8.9 %~~* 8.15 % of narrow range instrument span	≥~~15%~~ 10 %* of narrow range instrument span

* Loop design flow = ~~89,500~~ 85,000 gpm
** RTP = Rated Thermal Power

* Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)

TABLE 2.2-[illegible] (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
12. Steam/Feedwater Flow Mismatch Coincident With	Feed Flow ≤23.9% below rated Steam Flow	Feed Flow ≤20% below rated Steam Flow
Steam Generator Water Level-Low	≥~~13.2%~~ ~~8.9%~~+ $\boxed{8.15\%}$ of narrow range instrument span	≥~~15%~~ 10%+ of narrow range instrument span
13. Undervoltage - 4.16 kV Busses A and B	≥69% bus voltage	≥70% bus voltage
14. Underfrequency - Trip of Reactor Coolant Pump Breaker(s) Open	≥55.9 Hz	≥56.1 Hz
15. Turbine Trip		
a. Auto Stop Oil Pressure	≥42 psig	≥45 psig
b. Turbine Stop Valve Closure	Fully Closed***	Fully Closed***
16. Safety Injection Input from ESF	N. A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	≥6.0×10^{-11} amps	Nominal 1×10^{-10} amp

*** Limit switch is set when Turbine Stop Valves are fully closed.

* **Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)**

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

$$\Delta T \left\{\frac{1 + \tau_1 S}{1 + \tau_2 S}\right\} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_1 - K_2 \frac{(1 + \tau_4 S)}{(1 + \tau_5 S)} \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT = Measured ΔT by RTD Instrumentation

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead/Lag compensator on measured ΔT; τ_1=0s, τ_2 =0s

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT; τ_3 = 0s

ΔT_0 = Indicated ΔT at RATED THERMAL POWER

K_1 = ~~1.095~~; ~~1.25~~* 1.24

K_2 = ~~0.0107~~/°F; ~~0.016~~* 0.017

$\frac{1 + \tau_4 S}{1 + \tau_5 S}$ = The function generated by the lead-lag compensator for T_{avg} dynamic compensation;

τ_4, τ_5 = Time constants utilized in the lead-lag compensator for T_{avg}, τ_4 = 25s, τ_5 = 3 s;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg}; τ_6 = 0s

T' ≤ ~~574.2°F~~ 577.2 °F (Nominal T_{avg} at RATED THERMAL POWER);

K_3 = ~~0.000453~~/psig; ~~0.0011~~* 0.001

P = Pressurizer pressure, psig;

* Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P' $\geq$ 2235 psig (Nominal RCS operating pressure);

S = Laplace transform operator, s^{-1};

and f_1 (ΔI) is a function of the indicated difference between top and bottom detectors of the power-range neutron ion chambers; with gains to be selected based on measured instrument response during plant startup tests such that:

(1) For $q_t - q_b$ between ~~- 14%~~ ~~-46%~~* -50% and ~~+ 10%~~ +2%*, f_1 (ΔI) = 0, where q_t and q_b are percent RATED THERMAL POWER in the top and bottom halves of the core respectively, and $q_t + q_b$ is total THERMAL POWER in percent of RATED THERMAL POWER;

(2) For each percent that the magnitude of $q_t - q_b$ exceeds ~~- 14%~~ ~~-46%~~* -50%, the ΔT Trip Setpoint shall be automatically reduced by ~~1.5%~~ 0.0% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds ~~+ 10%~~ +2%*, the ΔT Trip Setpoint shall be automatically reduced by ~~1.5%~~ ~~2.3%~~* 2.19% of its value at RATED THERMAL POWER.

NOTE 2: The channels maximum trip setpoint shall not exceed its computed setpoint by more than ~~1.5%~~ ~~0.73%~~* 0.84% of instrument span.

* Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

$$\Delta T \left\{\frac{1 + \tau_1 S}{1 + \tau_2 S}\right\} \left(\frac{1}{1 + \tau_3 S}\right) \leq \Delta T_0 \left\{K_4 - K_5 \left(\frac{\tau_7 S}{1 + \tau_7 S}\right) \left(\frac{1}{1 + \tau_6 S}\right) T - K_6 \left[T \left(\frac{1}{1 + \tau_6 S}\right) - T''\right] - f_2\ (\Delta I)\right\}$$

Where:

ΔT = As defined in Note 1,

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = As defined in Note 1,

$\frac{1}{1 + \tau_3 S}$ = As defined in Note 1,

ΔT_0 = As defined in Note 1,

K_4 $\leq$ $\cancel{1.09}$, 1.10 *

K_5 $\geq$ 0.02/°F for increasing average temperature and 0 for decreasing average temperature,

$\frac{\tau_7 S}{1 + \tau_7 S}$ = The function generated by the rate-lag compensator for T_{avg} dynamic compensation,

τ_7 = Time constants utilized in the rate-lag compensator for T_{avg}, $\tau_7 \geq 10$ s,

$\frac{1}{1 + \tau_6 S}$ = As defined in Note 1,

* Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

~~0.00232~~* ← $\boxed{0.0016}$

NOTE 3: (Continued)

K_6	=	~~0.00068~~/°F for T > T"
	=	0 for T ≤ T",
T	=	As defined in Note 1,
T"	=	~~Indicated T_{avg} at RATED THERMAL POWER (Calibration temperature for ΔT instrumentation, ≤ 574.2°F)~~, (INSERT) ≤ 577.2°F (Nominal T_{avg} at RATED THERMAL POWER)
S	=	As defined in Note 1, and
f_2 (ΔI)	=	0 for all ΔI

NOTE 4: The channel's maximum trip setpoint shall not exceed its computed trip setpoint by more than ~~1.4%~~ of instrument span.

~~0.4%~~*

$\boxed{0.96\ \%}$

* Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)

3/4.2.2 HEAT FLUX HOT CHANNEL FACTOR - F_Q (Z)

LIMITING CONDITION FOR OPERATION

3.2.2 $F_Q^L(Z)$ shall be limited by the following relationships:

$$F_Q^M(Z) \leq \frac{[F_Q]^L}{P} \times [K(Z)] \text{ for } P > 0.5$$

$$F_Q^M(Z) \leq \frac{[F_Q]^L}{0.5} \times [K(Z)] \text{ for } P \leq 0.5$$

(INSERT) F_Q limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

where: $[F_Q]^L$ = ~~2.32 limit~~

$P = \dfrac{\text{Thermal Power}}{\text{Rated Thermal Power}}$,

$[F_Q]^M$ = The Measured Value, and

K(Z) for a given core height, is specified in the K(Z) curve, defined in the CORE OPERATING LIMITS REPORT.

APPLICABILITY: MODE 1

ACTION:

With the measured value of $F_Q^M(Z)$ exceeding its limit:

a. Reduce THERMAL POWER at least 1% for each 1% $F_Q^M(Z)$ exceeds $F_Q^L(Z)$ within 15 minutes and similarly reduce the Power Range Neutron Flux - High Trip Setpoints within the next 4 hours; POWER OPERATION may proceed for up to a total of 72 hours; subsequent POWER OPERATION may proceed provided the Overpower Delta-T Trip Setpoints (value of K_4) have been reduced at least 1% for each 1% $F_Q^M(Z)$ exceeds the $F_Q^L(Z)$; and

b. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced power limit required by ACTION a., above; THERMAL POWER may then be increased provided $F_Q^M(Z)$ is demonstrated through incore mapping to be within its limit.

POWER DISTRIBUTION LIMITS

3/4.2.3 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

LIMITING CONDITION FOR OPERATION

3.2.3 $F^N_{\Delta H}$ shall be limited by the following relationship:

$$F^N_{\Delta H} \le \underset{F_{\Delta H}^{RTP}}{\cancel{1.62}}\,[1.0 + \underset{PF_{\Delta H}}{\cancel{0.3}}\,(1-P)],$$

Where:

(INSERT)

$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT

$PF_{\Delta H}$ = Power Factor multiplier for $F_{\Delta H}$ as specified in the CORE OPERATING LIMITS REPORT

$$P = \frac{\text{THERMAL POWER}}{\text{RATED THERMAL POWER}}$$

APPLICABILITY: MODE 1.

ACTION:

With $F^N_{\Delta H}$ exceeding its limit:

a. Within 2 hours either:

1. Restore $F^N_{\Delta H}$ to within the above limit, or
2. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER and reduce the Power Range Neutron Flux - High Trip Setpoint to less than or equal to 55% of RATED THERMAL POWER within the next 4 hours.

b. Within 24 hours of initially being outside the above limit, verify through incore flux mapping that $F^N_{\Delta H}$ has been restored to within the above limit, or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 2 hours.

c. Identify and correct the cause of the out-of-limit condition prior to increasing THERMAL POWER above the reduced THERMAL POWER limit required by ACTION a.2. and/or b., above; subsequent POWER OPERATION may proceed provided that $F^N_{\Delta H}$ is demonstrated, through incore flux mapping, to be within the limit of acceptable operation prior to exceeding the following THERMAL POWER levels:

1. A nominal 50% of RATED THERMAL POWER,
2. A nominal 75% of RATED THERMAL POWER, and
3. Within 24 hours of attaining greater than or equal to 95% of RATED THERMAL POWER.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB-related parameters shall be maintained within the following limits:

a. Reactor Coolant System T_{avg} ≤ ~~576.6°F~~ 581.2

b. Pressurizer Pressure ≥ ~~2209~~ 2200 psig*, and

c. Reactor Coolant System Flow ≥ ~~277,900~~ 264,000 gpm

APPLICABILITY: MODE 1.

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

INSERT ATTACHMENT (HERE) (a)

SURVEILLANCE REQUIREMENTS

4.2.5.1 ~~Each of the parameters shown above shall be verified to be within its limits at least once per 12 hours.~~

3 4.2.5.2 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4 4.2.5.3 ~~The RCS flow rate shall be demonstrated by measurement once per 18 months.~~

INSERT ATTACHMENT (HERE) (a)

*Limit not applicable during either a THERMAL POWER ramp in excess of 5% of RATED THERMAL POWER per minute or a THERMAL POWER step in excess of 10% of RATED THERMAL POWER.

(a) Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)

INSERT TO TS SURVEILLANCE REQUIREMENTS 4.2.5.1 - 4.2.5.4

4.2.5.1 Reactor Coolant System T_{avg} and Pressurizer Pressure shall be verified to be within their limits at least once per 12 hours.

4.2.5.2 RCS flow rate shall be monitored for degradation at least once per 12 hours.

4.2.5.3 The RCS flow rate indicators shall be subjected to a CHANNEL CALIBRATION at least once per 18 months.

4.2.5.4 After each fuel loading, and at least once per 18 months, the RCS flow rate shall be determined by precision heat balance after exceeding 90% RATED THERMAL POWER. The measurement instrumentation shall be calibrated within 90 days prior to the performance of the calorimetric flow measurement. The provisions of 4.0.4 are not applicable for performing the precision heat balance flow measurement.

* Note to reviewer: The above insert is identical to the wording currently under review by the NRC for the Revised Thermal Design Procedure Technical Specification change (L-95-131 and L-95-250).

TABLE 3.3-3

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection (Reactor Trip, Turbine Trip, Feedwater Isolation, Control Room Ventilation Isolation, Start Diesel Generators, Containment Phase A Isolation (except Manual SI), Containment Cooling Fans, Containment Filter Fans, Start Sequencer, Component Cooling Water, Start Auxiliary Feedwater and Intake Cooling Water)		
a. Manual Initiation	N.A.	N.A.
b. Automatic Actuation Logic	N.A.	N.A.
c. Containment Pressure--High	$\leq$4.5 psig	$\leq$4.0 psig
d. Pressurizer Pressure--Low	$\geq$1712 psig	$\geq$1730 psig
e. High Differential Pressure Between the Steam Line Header and any Steam Line.	$\leq$114 psig	$\leq$100 psi
f. Steam Line Flow--High	$\leq$A function defined as follows: A ΔP corresponding to ~~42.6%~~ 44% steam flow at 0% load increasing linearly from 20% load to a value corresponding to ~~122.6%~~ steam flow at full 116.5% load.	$\leq$A function defined as follows: A ΔP corresponding to 40% steam flow at 0% load increasing linearly from 20% load to a value corresponding to ~~120 %~~ steam flow at full 114% load.

TABLE 3.3-[illegible] (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (Continued)		
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High Coincident with:	≤22.6 psig	≤20.0 psig
Containment Pressure--High	≤4.5 psig	≤4.0 psig
d. Steam Line Flow--High	≤A function defined as follows: A ΔP corresponding to ~~42.6%~~ 44% steam flow at 0% load increasing linearly from 20% load to a value corresponding to ~~122.6%~~ 116.5% steam flow at full load.	≤A function defined follows: A ΔP corresponding to 40% steam flow at 0% load increasing linearly from 20% load to a value corresponding to ~~120%~~ 114% steam flow at full load.
Coincident with: Steam Line Pressure--Low	≥588 psig	≥614 psig
or T_{avg}--Low	≥542.5°F	≥543°F
5. Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
5. Feedwater Isolation (Continued)		
c. Steam Generator Water Level High-High	≤81.9% of narrow range instrument span	≤80% of narrow range instrument span
6. Auxiliary Feedwater (3)		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level--Low-Low	~~≤13%~~ ~~≥8.9%~~* 8.15% of narrow range instrument span.	~~≥15%~~ 10%* of narrow range instrument span.
c. Safety Injection	See Item 1. above for all Safety Injection Allowable Values.	See Item 1: above for all Safety Injection Trip Setpoints.
d. Bus Stripping	See Item 7. below for all Bus Stripping Allowable Values.	See Item 7. below for all Bus Stripping Trip Setpoints.
e. Trip of All Main Feedwater Pump Breakers	N.A.	N.A.
7. Loss of Power		
a. 4.16 kV Busses A and B (Loss of Voltage)	N.A.	N.A.

* Revised Thermal Design Procedure (RTDP) Technical Specification change under review by the NRC (L-95-131 and L-95-250)

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE* with a lift setting of 2485 psig ± 1%.** *** +2%, -3%

APPLICABILITY: MODES 4 and 5.

ACTION:

With no pressurizer Code safety valve OPERABLE, immediately suspend all operations involving positive reactivity changes and place an OPERABLE RHR loop into operation in the shutdown cooling mode.

SURVEILLANCE REQUIREMENTS

4.4.2.1 No additional requirements other than those required by Specification 4.0.5.

(INSERT)

*** All valves tested must have "as-left" lift setpoints that are within ± 1% of the lift setting value.

*While in MODE 5, an equivalent size vent pathway may be used provided that the vent pathway is not isolated or sealed.
**The lift setting pressure shall correspond to ambient conditions of the valve at nominal operating temperature and pressure.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 2485 psig ± 1%* ** + 2%, - 3%

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

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

SURVEILLANCE REQUIREMENTS

4.4.2.2 No additional requirements other than those required by Specification 4.0.5.

** All valves tested must have "as-left" lift setpoints that are within ± 1% of the lift setting value.

(INSERT)

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

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F

SERVICE PERIOD: 20 EFPY (19) — RT_{NDT} @ 1/4 THICKNESS = 252.5°F

HEATUP RATES: UP TO 60°F/HR — RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

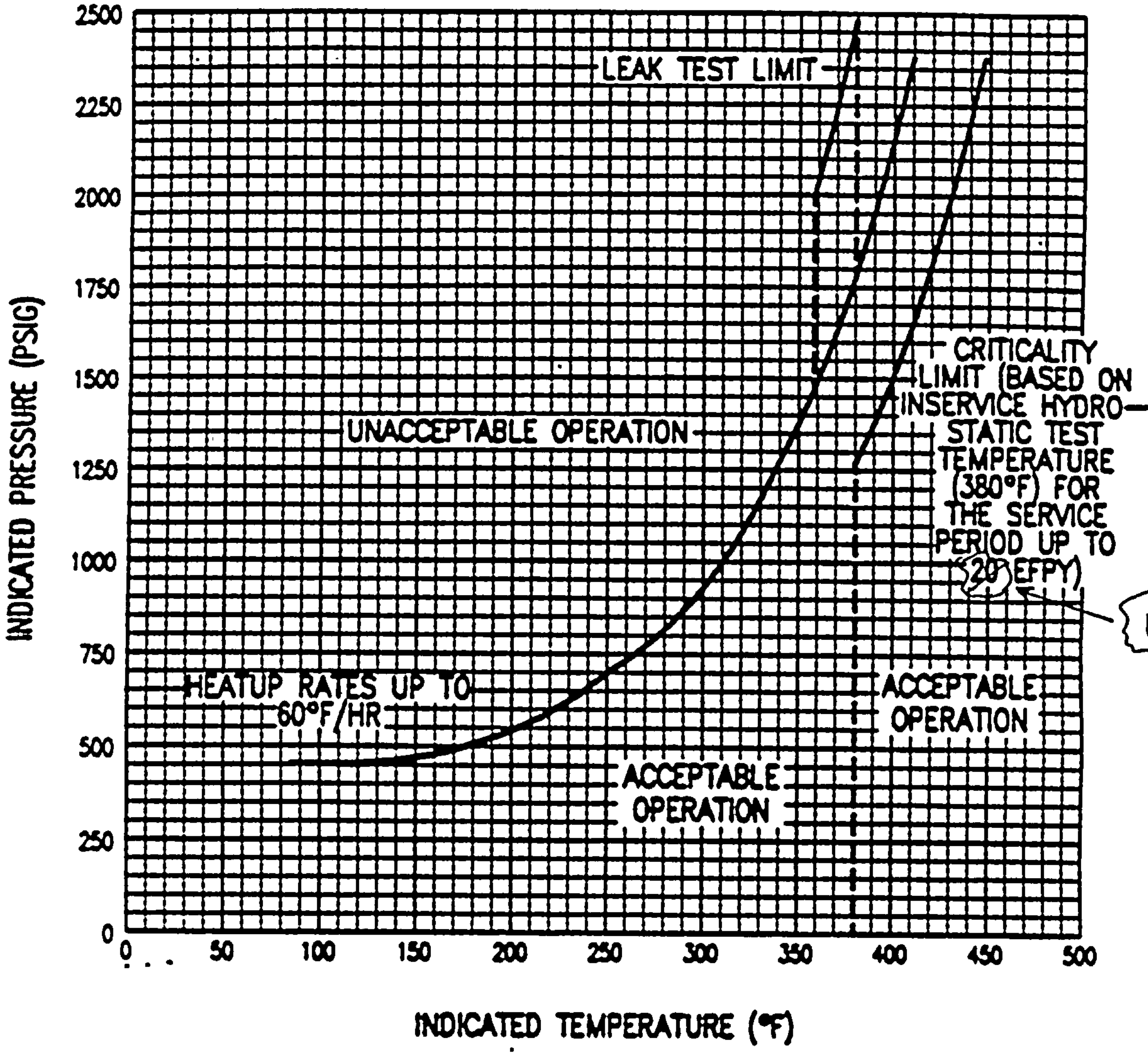


FIGURE 3.4-2

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (60°F/hr) - APPLICABLE UP TO 20 EFPY

19

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F

19

SERVICE PERIOD: 20 EFPY — RT_{NDT} @ 1/4 THICKNESS = 252.5°F

HEATUP RATES: UP TO 100°F/HR — RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

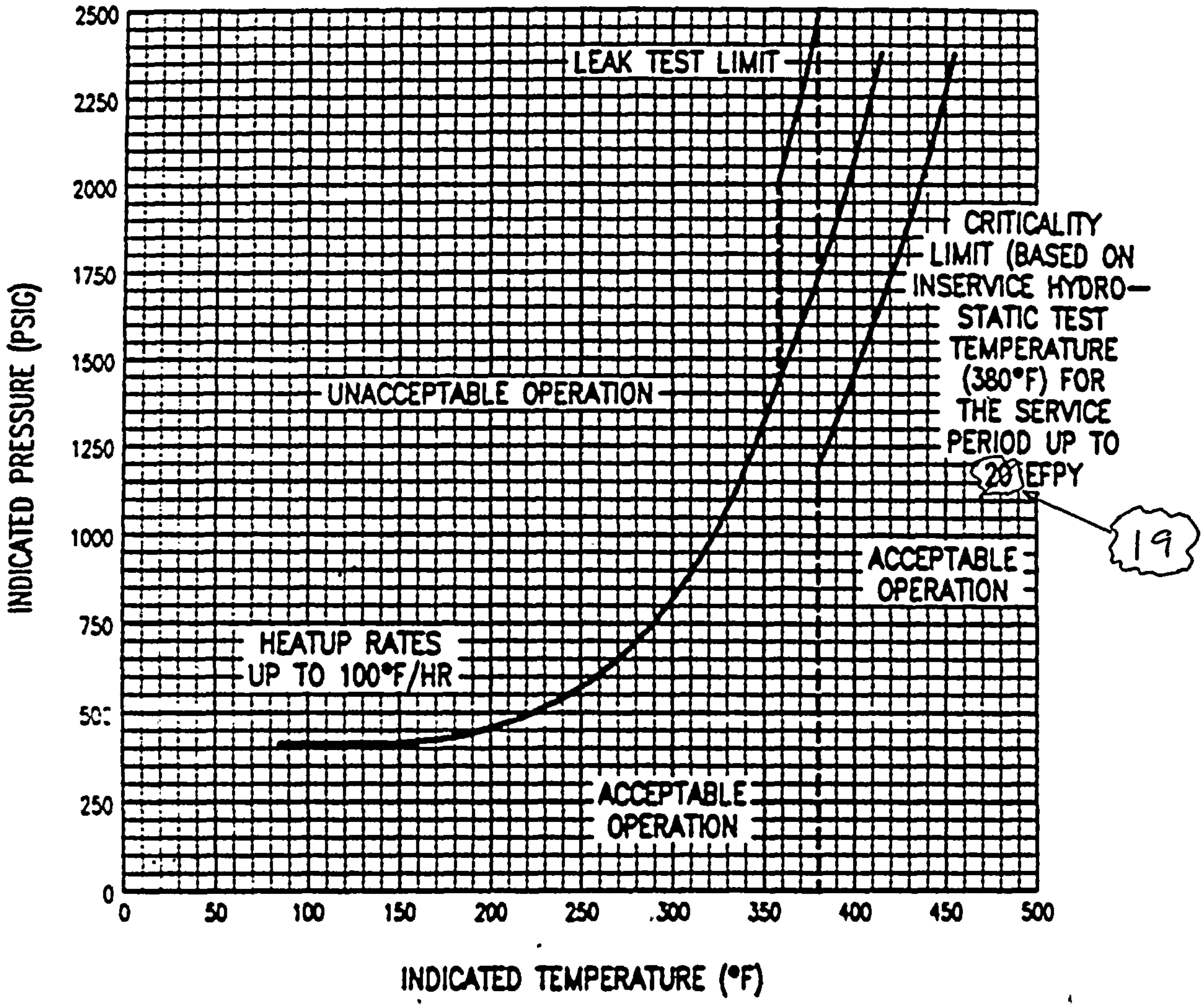


FIGURE 3.4-3

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM HEATUP LIMITATIONS (100°F/hr) - APPLICABLE UP TO 20 EFPY

19

MATERIAL PROPERTY BASIS

CONTROLLING MATERIAL: CIRCUMFERENTIAL WELD
INITIAL RT_{NDT}: 10°F
SERVICE PERIOD: 20 EFPY 19
COOLDOWN RATES: UP TO 100°F/HR

RT_{NDT} @ 1/4 THICKNESS = 252.5°F
RT_{NDT} @ 3/4 THICKNESS = 200.4°F

NOTE: NO MARGINS ARE GIVEN FOR POSSIBLE INSTRUMENT ERRORS.

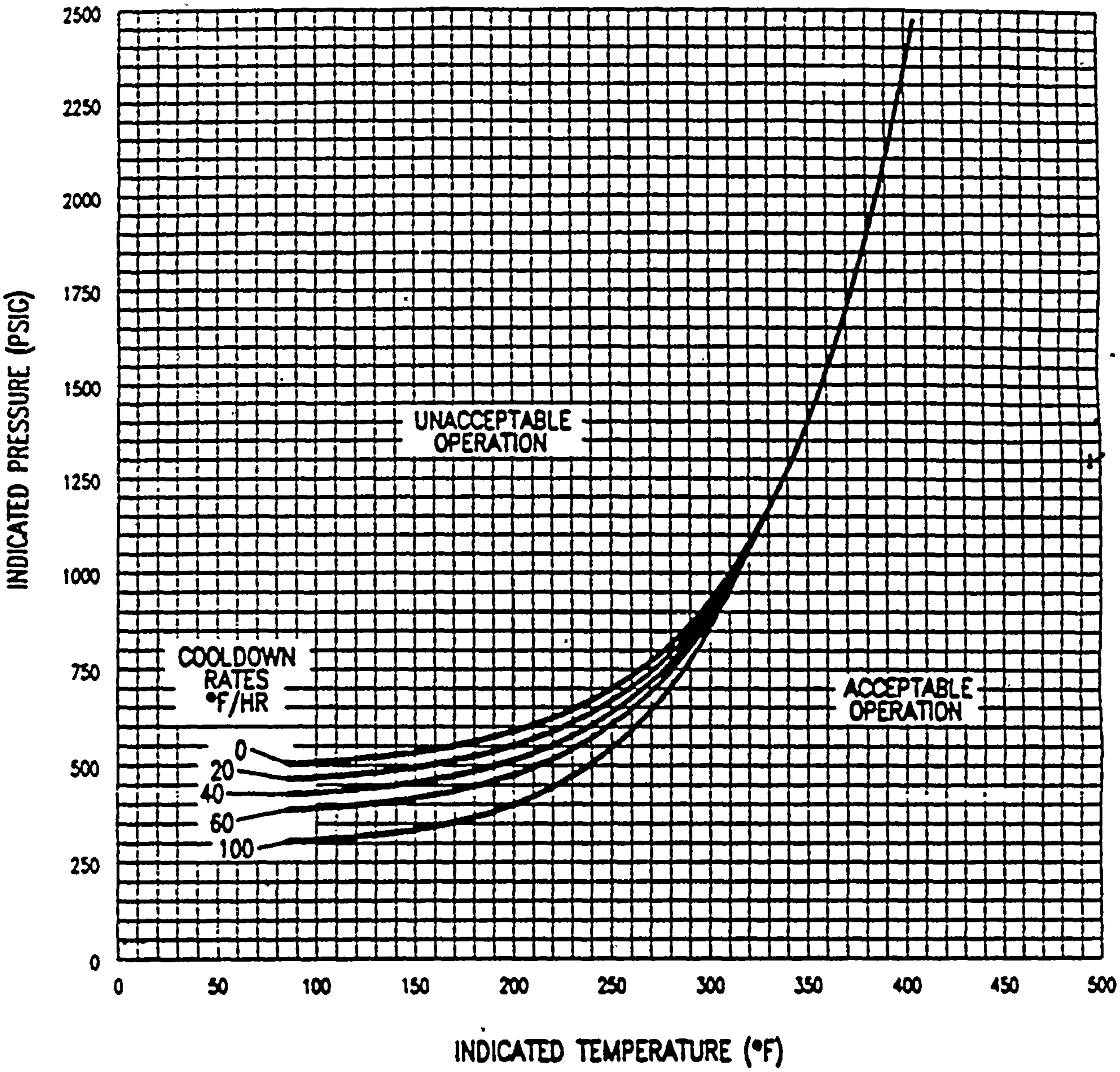


FIGURE 3.4-4

TURKEY POINT UNITS 3 & 4

REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS (100°F/hr) - APPLICABLE UP TO 20 EFPY 19

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS component and flow path shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying by control room indication that the following valves are in the indicated positions with power to the valve operators removed:

Valve Number	Valve Function	Valve Position
864A and B	Supply from RWST to ECCS	Open
862A and B	RWST Supply to RHR pumps	Open
863A and B	RHR Recirculation	Closed
866A and B	H.H.S.I. to Hot Legs	Closed
HCV-758*	RHR HX Outlet	Open

To permit temporary operation of these valves for surveillance or maintenance purposes, power may be restored to these valves for a period not to exceed 24 hours.

b. At least once per 31 days by:

1) Verifying that the ECCS piping is full of water by venting the ECCS pump casings and accessible discharge piping,

2) Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position, and

3) Verifying that each RHR Pump develops the indicated differential pressure applicable to the operating conditions in accordance with Figure 3.5-1 when tested pursuant to Specification 4.0.5.

c. At least once per 92 days by:

1) Verifying that each SI pump develops the indicated differential pressure applicable to the operating conditions when tested pursuant to Specification 4.0.5.

1083 SI pump $\geq$ ~~1126~~ psid at a metered flowrate $\geq$ 300 gpm (normal alignment or Unit 4 SI pumps aligned to Unit 3 RWST), or

1113 $\geq$ ~~1156~~ psid at a metered flowrate $\geq$ 280 gpm (Unit 3 SI pumps aligned to Unit 4 RWST).

*Air Supply to HCV-758 shall be verified shut off and sealed closed once per 31 days.

CONTAINMENT SYSTEMS

EMERGENCY CONTAINMENT COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.2.2 Three emergency containment cooling units shall be OPERABLE.

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

ACTION:

a. With one of the above required emergency containment cooling units inoperable restore the inoperable cooling unit to OPERABLE status ithin 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With two or more of the above required emergency containment cooling units inoperable, restore at least two cooling units to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore all of the above required cooling units to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.2 Each emergency containment cooling unit shall be demonstrated OPERABLE:

a. At least once per 31 days by starting each cooler unit from the control room and verifying that each unit motor reaches the nominal operating current for the test conditions and operates for at least 15 minutes.

b. At least once per 18 months by:

(INSERT)

1) Verifying that ~~each unit~~ starts automatically on a safety injection (SI) test signal, and

2) Verifying a cooling water flow rate of greater than or equal to 2000 gpm to each cooler.

two emergency containment cooling units

TABLE 3.7-1

MAXIMUM ALLOWABLE POWER LEVEL WITH INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION

MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR	MAXIMUM ALLOWABLE POWER LEVEL (PERCENT OF RATED THERMAL POWER)
1	~~56~~ ← 53
2	~~35~~ ← 33
3	14

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

VALVE NUMBER	Loop A	Loop B	Loop C	LIFT SETTING (~~±1%~~ ± 3 %)* **	ORIFICE SIZE SQUARE INCHES
1.	RV1400	RV1405	RV1410	1085 psig	16
2.	RV1401	RV1406	RV1411	1100 psig	16
3.	RV1402	RV1407	RV1412	1115 psig	16
4.	RV1403	RV1408	RV1413	1130 psig	16

** All valves tested must have "as-left" lift setpoints that are within ± 1% of the Lift Setting value listed in Table 3.7-2.

(INSERT)

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

PLANT SYSTEMS

CONDENSATE STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.7.1.3 The condensate storage tanks (CST) system shall be OPERABLE with:

Opposite Unit in MODES 4, 5 or 6

~~An~~ A minimum indicated water volume of ~~185,000~~ 210,000 gallons in either or both condensate storage tanks.

Opposite Unit in MODES 1, 2 or 3

~~An~~ A minimum indicated water volume of ~~370,000~~ 420,000 gallons.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

Opposite Unit in MODES 4, 5 or 6

With the CST system inoperable, within 4 hours restore the CST system to OPERABLE status or be in at least HOT STANDBY in the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

Opposite Unit in MODES 1, 2 or 3

1) With the CST system inoperable due to ~~containing~~ indicating less than ~~370,000~~ 420,000 gallons, but greater than or equal to ~~185,000~~ 210,000 gallons indicated, within 4 hours restore the inoperable CST system to OPERABLE status or place one unit in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

2) With the CST system inoperable with less than ~~185,000~~ 210,000 gallons indicated, within 1 hour restore the CST system to OPERABLE status or be in at least HOT STANDBY within the next 12 hours and in HOT SHUTDOWN within the following 6 hours. This ACTION applies to both units simultaneously.

SURVEILLANCE REQUIREMENTS (Continued)

4.7.1.3 The condensate storage tank (CST) system shall be demonstrated OPERABLE at least once per 12 hours by verifying the ~~contained~~ indicated water volume is within its limit when the tank is the supply source for the auxiliary feedwater pumps.

PLANT SYSTEMS

STANDBY STEAM GENERATOR FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

135,000

3.7.1.6 Two Standby Steam Generator Feedwater Pumps shall be OPERABLE* and at least ~~60,000~~ gallons of water ~~(available volume)~~, shall be in the Demineralized Water Storage Tank**.

(insert)

(indicated volume)

APPLICABILITY: MODES 1, 2 and 3

ACTION:

a. With one Standby Steam Generator Feedwater Pump inoperable, restore the inoperable pump to available status within 30 days or submit a SPECIAL REPORT per 3.7.1.6d.

b. With both Standby Steam Generator Feedwater Pumps inoperable, restore at least one pump to OPERABLE status within 24 hours, or:

   1. Notify the NRC within the following 4 hours, and provide cause for the inoperability and plans to restore pump(s) to OPERABLE status and,

   2. Submit a SPECIAL REPORT per 3.7.1.6d.

indicated

c. With less than ~~60,000~~ 135,000 gallons of water in the Demineralized Water Storage Tank restore the available volume to at least ~~60,000~~ 135,000 gallons within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.

d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the inoperability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2.

SURVEILLANCE REQUIREMENTS

4.7.1.6.1 The Demineralized Water Storage tank water volume shall be determined to be within limits at least once per 24 hours.

4.7.1.6.2 At least monthly verify the Standby Steam Generator Feedwater Pumps are OPERABLE by testing in recirculation on a STAGGERED TEST BASIS.

4.7.1.6.3 At least once per 18 months, verify operability of the respective standby steam generator feedwater pump by starting each pump and providing feedwater to the steam generators.

*These pumps do not require plant safety related emergency power sources for operability and the flowpath is normally isolated.

**The Demineralized Water Storage Tank is non-safety grade.

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

1) Verifying that the air cleanup system satisfies the in-place penetration and bypass leakage testing acceptance criteria of greater than or equal to 99% DOP and halogenated hydrocarbon removal at a system flow rate of 1000 cfm ±10%.

2) Verifying, within 31 days after removal, that ~~a~~ laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, and analyzed per ANSI N510-1975, meets the criteria for methyl iodine removal efficiency of greater than or equal to 99% ~~90%~~ or the charcoal be replaced with charcoal that meets or exceeds the criteria of position C.6.a. of Regulatory Guide 1.52 (Revision 2), and

3) Verifying by a visual inspection the absence of foreign materials and gasket deterioration.

d. At least once per 12 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches Water Gauge while operating the system at a flow rate of 1000 cfm ±10%;

e. At least once per 18 months by verifying that on a Containment Phase "A" Isolation test signal the system automatically switches into the recirculation mode of operation.

PEAKING FACTOR LIMIT REPORT (Continued)

Factor Limit Report, the Peaking Factor Limit Report shall be provided to the NRC Document Control desk with copies to the Regional Administrator and the Resident Inspector within 30 days of their implementation, unless otherwise approved by the Commission.

The analytical methods used to generate the Peaking Factor limits shall be those previously reviewed and approved by the NRC. If changes to these methods are deemed necessary they will be evaluated in accordance with 10 CFR 50.59 and submitted to the NRC for review and approval prior to their use if the change is determined to involve an unreviewed safety question or if such a change would require amendment of previously submitted documentation.

CORE OPERATING LIMITS REPORT

6.9.1.7 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle for the following:

1. Axial Flux Difference for Specifications 3.2.1.

2. Control Rod Insertion Limits for Specification 3.1.3.6.

3. Heat Flux Hot Channel Factor - $F_Q(Z)$ for Specification 3/4.2.2.

4. Nuclear Enthalpy Rise Hot Channel Factor for Specification 3/4.2.3

The analytical methods used to determine the AFD limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-10216-P-A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," June 1983.

2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974.

$F_Q(z)$, $F_{\Delta H}$ and

The analytical methods used to determine the K(Z) curve shall be those previously reviewed and approved by the NRC in:

1. WCAP-9220-P-A, Rev. 1, "Westinghouse ECCS Evaluation Model - 1981 Version," February 1982.

2. WCAP-9561-P-A, ADD. 3, Rev. 1, "BART A-1: A Computer Code for the Best Estimate Analysis of Reflood Transients - Special Report: Thimble Modeling W ECCS Evaluation Model."

INSERT ATTACHMENT (HERE)

The analytical methods used to determine the Rod Bank Insertion Limits shall be those previously reviewed and approved by the NRC in:

1. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985.

The ability to calculate the COLR nuclear design parameters are demonstrated in:

1. Florida Power & Light Company Topical Report NF-TR-95-01, "Nuclear Physics Methodology for Reload Design of Turkey Point & St. Lucie Nuclear Plants".

INSERT TO TS 6.9.1.7

[NOTE: References 3 and 4 are included in the Proposed License Amendments for the Small Break Loss-of-Coolant (SBLOCA) Reanalysis, as transmitted to the NRC via L-95-193.]

5. WCAP-10266-P-A, Rev. 2 (proprietary) and WCAP-11524-NP-A, Rev. 2 (non-proprietary), "The 1981 Version of the Westinghouse ECCS Evaluation Model Using the BASH Code," May 1988.

6. NTD-NRC-94-4143, "Change in Methodology for Execution of BASH Evaluation Model," May 23, 1994.

CORE OPERATING LIMITS REPORT (Continued

Topical Report NF-TR-95-01 was approved by the NRC for use by Florida Power & Light Company in:

1. Safety Evaluation by the Office of Nuclear Reactor Regulations Related to Amendment No. 174 to Facility Operating License DPR-31 and Amendment No. 168 to Facility Operating License No. DPR-41, Florida Power & Light Company Turkey Point Units Nos. 3 and 4, Docket Nos. 50-250 and 50-251.

$F_Q(z)$, $F_{\Delta H}$

The AFD, K(Z), and Rod Bank Insertion Limits shall be determined such that all applicable limits of the safety analyses are met. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be provided upon issuance, for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and the Resident Inspector, unless otherwise approved by the Commission.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report as stated in the Specifications within Sections 3.0, 4.0, or 5.0.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level;

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety;

c. All REPORTABLE EVENTS;

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications;

e. Records of changes made to the procedures required by Specification 6.8.1;

f. Records of radioactive shipments;

g. Records of sealed source and fission detector leak tests and results; and

h. Records of annual physical inventory of all sealed source material of record.

6.10.3 The following records shall be retained for the duration of the unit Operating License:

2.1 SAFETY LIMITS

BASES

2.1.1 REACTOR CORE

The restrictions of this Safety Limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excessive cladding temperatures because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore THERMAL POWER and reactor coolant temperature and pressure have been related to DNB. This relationship has been developed to predict the DNB flux and the location of DNB for axially uniform and nonuniform heat flux distributions. The local DNB heat flux ratio (DNBR) is defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux and is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability with 95 percent confidence that the minimum DNBR of the limiting rod during Condition I and II events is greater than or equal to the DNBR limit of the DNB correlation being used. The correlation DNBR limit is established based on the entire applicable experimental data set such that there is a 95 percent probability with 95 percent confidence that DNB will not occur when the minimum DNBR is at the DNBR limit.

The curves of Figure 2.1-1 show the loci of points of THERMAL POWER, Reactor Coolant System pressure and average temperature for which the minimum DNBR is no less than the design DNBR value, or the average enthalpy at the vessel exit is equal to the enthalpy of saturated liquid.

These curves are based on an enthalpy hot channel factor, $F^N_{\Delta H}$ ~~of 1.62~~ and and a reference cosine with a peak of 1.55 for axial power shape. An allowance is included for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

$F^{RTP}_{\Delta H}$. $$F^N_{\Delta H} \le \cancel{1.62}[1+\cancel{0.3}(1-P)]$$ $PF_{\Delta H}$

Where P is the fraction of RATED THERMAL POWER.

INSERT ATTACHMENT (HERE)

These limiting heat flux conditions are higher than those calculated for the range of all control rods fully withdrawn to the maximum allowable control rod insertion limit assuming the axial power imbalance is within the limits of the f (ΔI) function of the Overtemperature trip. When the axial power imbalance is not within the tolerance, the axial power imbalance effect on the Overtemperature ΔT trips will reduce the setpoints to provide protection consistent with core Safety Limits.

INSERT TO TS BASES 2.1.1

$F_{\Delta H}^{RTP}$ = $F_{\Delta H}$ limit at RATED THERMAL POWER as specified in the CORE OPERATING LIMITS REPORT.

$PF_{\Delta H}$ = Power Factor multiplier for $F_{\Delta H}$ as specified in the CORE OPERATING LIMITS REPORT.

LIMITING SAFETY SYSTEM SETTINGS

BASES

Undervoltage and - 4.16 kV Bus A and B Trips (Continued)

power the Undervoltage Bus trips are automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Turbine Trip

A Turbine trip initiates a Reactor trip. On decreasing power, the Reactor Trip from the Turbine trip is automatically blocked by P-7 (a power level of approximately 10% of RATED THERMAL POWER with a turbine first stage pressure at approximately 10% of full power equivalent); and on increasing power, reinstated automatically by P-7.

Safety Injection Input from ESF

If a Reactor trip has not already been generated by the Reactor Trip System instrumentation, the ESF automatic actuation logic channels will initiate a Reactor trip upon any signal which initiates a Safety Injection. The ESF instrumentation channels which initiate a Safety Injection signal are shown in Table 3.3-3.

Reactor Coolant Pump Breaker Position Trip

The Reactor Coolant Pump Breaker Position Trips are anticipatory trips which provide reactor core protection against DNB. The open/close position trips assure a reactor trip signal is generated before the low flow trip setpoint is reached. ~~No credit was taken in the accident analyses for operation of these trips.~~ Their functional capability at the open/close position settings is required to enhance the overall reliability of the Reactor Protection System. Above P-7 (a power level of approximately 10% of RATED THERMAL POWER or a turbine first stage pressure at approximately 10% of full power equivalent) an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened. Above P-8 (a power level of approximately 45% of RATED THERMAL POWER) an automatic reactor trip will occur if one reactor coolant pump breaker is opened. On decreasing power between P-8 and P-7, an automatic reactor trip will occur if more than one reactor coolant pump breaker is opened and below P-7 the trip function is automatically blocked.

Underfrequency sensors are also installed on the 4.16 kV busses to detect underfrequency and initiate breaker trip on underfrequency. The underfrequency trip setpoints preserve the coast down energy of the reactor coolant pumps, in case of a grid frequency decrease so DNB does not occur.

3/4.2 POWER DISTRIBUTION LIMITS

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (1) maintaining the minimum DNBR in the core greater than or equal to the applicable design limit during normal operation and in short-term transients, and (2) limiting the fission gas release, fuel pellet temperature, and cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Condition I events provides assurance that the initial conditions assumed for the LOCA analyses are met and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of certain hot channel and peaking factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods;

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power; and

$F_{xy}(Z)$ Radial Peaking Factor, is defined as the ratio of peak power density to average power density in the horizontal plane at core elevation Z.

3/4.2.1 AXIAL FLUX DIFFERENCE

limit defined in the CORE OPERATING LIMITS REPORT

The limits on AXIAL FLUX DIFFERENCE (AFD) assure that the $F_Q(Z)$ ~~upper bound envelope of 2.32~~ times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full-length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady-state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

3/4.2.2 and 3/4.2.3 HEAT FLUX HOT CHANNEL FACTOR AND NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR

The limits on heat flux hot channel factor and nuclear enthalpy rise hot channel factor ensure that: (1) the design limits on peak local power density and minimum DNBR are not exceeded and (2) in the event of a LOCA the peak fuel clad temperature will not exceed the 2200°F ECCS acceptance criteria limit. The LOCA peak fuel clad temperature limit may be sensitive to the number of steam generator tubes plugged. ~~The current limit is valid for tube plugging levels up to 5%.~~

$F_Q(Z)$, Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux.

$F_{\Delta H}^{N}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

Each of these is measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to ensure that the limits are maintained provided:

a. Control rods in a single group move together with no individual rod insertion differing by more than ± 12 steps, indicated, from the group demand position;

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.6;

c. The control rod insertion limits of Specifications 3.1.3.5 and 3.1.3.6 are maintained; and

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE, is maintained within the limits.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. Five percent is the appropriate allowance for a full core map taken with the movable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance. These uncertainties only apply if the map is taken for purposes other than the determination of P_{BL} and P_{RB}.

$F_{\Delta H}^{N}$ will be maintained within its limits provided Conditions a. through d. above are maintained.

In the specified limit of $F_{\Delta H}^{N}$, there is an 8 percent allowance for uncertainties which means that normal operation of the core is expected to result in $F_{\Delta H}^{N} \leq \cancel{1.62}/1.08$. The logic behind the larger uncertainty in this

$F_{\Delta H}^{RTP}$ INSERT ATTACHMENT HERE

INSERT TO TS BASES 3/4.2.2 AND 3/4.2.3

, where $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}^{N}$ limit at RATED THERMAL POWER (RTP) specified in the CORE OPERATING LIMITS REPORT.

3/4.2.4 QUADRANT POWER TILT RATIO

The QUADRANT POWER TILT RATIO limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during STARTUP testing and periodically during power operation.

The limit of 1.02, at which corrective action is required, provides DNB and linear heat generation rate protection with x-y plane power tilts. A limit of 1.02 was selected to provide an allowance for the uncertainty associated with the indicated power tilt.

The 2-hour time allowance for operation with a tilt condition greater than 1.02 but less than 1.09 is provided to allow identification and correction of a dropped or misaligned control rod. In the event such action action does not correct the tilt, the margin for uncertainty on F_Q (Z) is reinstated by reducing the maximum allowed power by 3% for each percent of tilt in excess of 1.

For purposes of monitoring QUADRANT POWER TILT RATIO when one excore detector is inoperable, the movable incore detectors or incore thermocouple map are used to confirm that the normalized symmetric power distribution is consistent with the QUADRANT POWER TILT RATIO. The incore detector monitoring is done with a full incore flux map or two sets of four symmetric thimbles. The two sets of four symmetric thimbles is a unique set of eight detector locations. These locations are C-8, E-5, E-11, H-3, H-13, L-5, L-11, N-8.

3/4.2.5 DNB PARAMETERS

The limits on the DNB-related parameters assure that each of the parameters are maintained within the normal steady-state envelope of operation assumed in the transient and accident analyses. The limits are consistent with the initial FSAR assumptions and have been analytically demonstrated adequate to maintain a minimum DNBR above the applicable design limits throughout each analyzed transient. The indicated T_{avg} value of ~~576.6°F~~ 581.2°F and the indicated pressurizer pressure value of ~~2209~~ 2200 psig correspond to analytical limits of ~~578.2~~ 583.2°F and ~~2185~~ 2175 psig respectively, with allowance for measurement uncertainty.

The ~~indicated~~ measured RCS flow value of ~~277,900~~ 264,000 gpm corresponds to an analytical limit of ~~268,500~~ 255,000 gpm which is assumed to have a 3.5% calorimetric measurement uncertainty. ~~The above measurement uncertainty estimates assume that these instrument channel outputs are averaged to minimize the uncertainty.~~

The 12-hour periodic surveillance of these parameters through instrument readout is sufficient to ensure that the parameters are restored within their limits following load changes and other expected transient operation.

INSERT ATTACHMENT (HERE)

INSERT TO TS BASES 3/4.2.5

The 18-month periodic measurement of the RCS total flow rate is adequate to ensure that the DNB-related flow assumption is met and to ensure correlation of the flow indication channels with measured flow. Six month drift effects have been included for feedwater temperature, feedwater flow, steam pressure, and the pressurizer pressure inputs. The flow measurement is performed within ninety days of completing the cross-calibration of the hot leg and cold leg narrow range RTDs. The indicated percent flow surveillance on a 12-hour basis will provide sufficient verification that flow degradation has not occurred. ~~A change in indicated percent flow which is greater than the instrument channel inaccuracies and parallax errors is an appropriate indication of RCS flow degradation.~~

(INSERT)

An indicated percent flow which is greater than the thermal design flow plus instrument channel inaccuracies and parallax errors is acceptable for the 12 hour surveillance on RCS flow. To minimize measurement uncertainties it is assumed that the RCS flow channel outputs are averaged.

* Note to reviewer: The above insert was originally submitted to the NRC for review of the Revised Thermal Design Procedure Technical Specification change (L-95-131). The marked-up changes represent the subsequent proposed changes in accordance with the Uprate submittal.

3/4.4.2 SAFETY VALVES

The pressurizer Code safety valves operate to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. Each safety valve is designed to relieve 293,330 lbs per hour of saturated steam at the valve Setpoint. The relief capacity of a single safety valve is adequate to relieve any overpressure condition which could occur during shutdown. In the event that no safety valves are OPERABLE, an RCS vent opening of at least 2.50 square inches will provide overpressure relief capability and will prevent RCS overpressurization. In addition, the Overpressure Mitigating System provides a diverse means of protection against RCS overpressurization at low temperatures.

During operation, all pressurizer Code safety valves must be OPERABLE to prevent the RCS from being pressurized above its Safety Limit of 2735 psig. The combined relief capacity of all of these valves is greater than the maximum surge rate resulting from a complete loss-of-load assuming no Reactor trip until the first Reactor Trip System Trip Setpoint is reached (i.e., no credit is taken for a direct Reactor trip on the loss-of-load) and also assuming no operation of the power-operated relief valves or steam dump valves.

In Mode 5 only one pressurizer code safety is required for overpressure protection. In lieu of an actual operable code safety valve, an unisolated and unsealed vent pathway (i.e., a direct, unimpaired opening, a vent pathway with valves locked open and/or power removed and locked on an open valve) of equivalent size can be taken credit for as synonymous with an OPERABLE code safety.

Demonstration of the safety valves' lift settings will occur only during shutdown and will be performed in accordance with the provisions of Section XI of the ASME Boiler and Pressure Code.

(INSERT)

3/4.4.3 PRESSURIZER

The 12-hour periodic surveillance is sufficient to ensure that the maximum water volume parameter is restored to within its limit following expected transient operation. The maximum water volume (1133 cubic feet) ensures that a steam bubble is formed and thus the RCS is not a hydraulically solid system. The requirement that both backup pressurizer heater groups be OPERABLE enhances the capability of the plant to control Reactor Coolant System pressure and establish natural circulation.

The pressurizer code safety valves' lift settings allows a +2%, -3% setpoint tolerance for OPERABILITY; however, the valves are reset to within ±1% during the surveillance to allow for drift.

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 to 3.4-4 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 to 3.4-4 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 1983 Edition of Section III 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 GTSD-A-1.12, "Procedure for Developing Heatup and Cooldown 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 20 effective full power years (EFPY) of service life. The 20 EFPY service life period is chosen such that the limiting RT_{NDT} at the 1/4T location in

19 19

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, 3.4-3 and 3.4-4 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). 19

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, 3.4-3, and 3.4-4 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 50 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. 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

CONTAINMENT VENTILATION SYSTEM (Continued)

resilient material seal degradation and will allow opportunity for repair before gross leakage failures could develop. The 0.60 L_a leakage limit of Specification 3.6.1.2b. shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.2 DEPRESSURIZATION AND COOLING SYSTEMS

3/4.6.2.1 CONTAINMENT SPRAY SYSTEM

The OPERABILITY of the Containment Spray System ensures that containment depressurization capability will be available in the event of a LOCA. The pressure reduction and resultant lower containment leakage rate are consistent with the assumptions used in the safety analyses.

The allowable out-of-service time requirements for the Containment Spray System have been maintained consistent with that assigned other inoperable ESF equipment and do not reflect the additional redundancy in cooling capability provided by the Emergency Containment Cooling System. Pump performance requirements are obtained from the accidents analysis assumptions.

3/4.6.2.2 EMERGENCY CONTAINMENT COOLING SYSTEM

INSERT ATTACHMENT (HERE)

The OPERABILITY of the Emergency Containment Cooling System ensures that adequate heat removal capacity is available during post-LOCA conditions. The emergency containment coolers are a full capacity system and are redundant to the spray system in terms of heat removal function for design basis accident.

The allowable out-of-service time requirements for the Containment Cooling System have been maintained consistent with that assigned other inoperable ESF equipment and do not reflect the additional redundancy in cooling capability provided by the Containment Spray System.

The surveillance requirement for ECC flow is verified by correlating the test configuration value with the design basis assumptions for system configuration and flow. An 18-month surveilance interval is acceptable based on the use of water from the CCW system, which results in a low risk of heat exchanger tube fouling.

3/4.6.3 EMERGENCY CONTAINMENT FILTERING SYSTEM

The OPERABILITY of the Emergency Containment Filtering System ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting SITE BOUNDARY radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. System components are not subject to rapid deterioration. Visual inspection and operating/performance tests after maintenance, prolonged operation, and at the required frequencies provide assurances of system reliability and will prevent system failure. Filter performance tests are conducted in accordance with the methodology and intent of ANSI N510- 1975.

INSERT TO TS BASES 3/4.6.2.2

The OPERABILITY of the Emergency Containment Cooling (ECC) System ensures that the heat removal capacity is maintained with acceptable ranges following postulated design basis accidents. To support both containment integrity safety analyses and component cooling water system thermal analyses, a maximum of two ECCs can receive an automatic start signal following generation of a safety injection (SI) signal (one ECC receives an "A" train SI signal and another ECC receives a "B" train SI signal). To support post-LOCA long-term containment pressure/temperature analyses, a maximum of two ECCs are required to operate. The third (swing) ECC is required to be OPERABLE to support manual starting following a postulated LOCA event for containment pressure/temperature suppression.

SAFETY VALVES (Continued)

Operation with less than all four MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

3/4.7.1.2 AUXILIARY FEEDWATER SYSTEM INSERT ATTACHMENT HERE

The OPERABILITY of the Auxiliary Feedwater System ensures that the Reactor Coolant System can be cooled down to less than 350°F from normal operating conditions in the event of a total loss-of-offsite power. Steam can be supplied to the pump turbines from either or both units through redundant steam headers. Two D.C. motor operated valves and one A.C. motor operated valve on each unit isolate the three main steam lines from these headers. Both the D.C. and A.C. motor operated valves are powered from safety-related sources. Auxiliary feedwater can be supplied through redundant lines to the safety-related portions of the main feedwater lines to each of the steam generators. Air operated fail closed flow control valves are provided to modulate the flow to each steam generator. Each steam driven auxiliary feedwater pump has sufficient capacity for single and two unit operation to ensure that adequate feedwater flow is available to remove decay heat and reduce the Reactor Coolant System temperature to less than 350°F when the Residual Heat Removal System may be placed into operation.

ACTION statement 2 describes the actions to be taken when both auxiliary feedwater trains are inoperable. The requirement to verify the availability of both standby feedwater pumps is to be accomplished by verifying that both pumps have successfully passed their monthly surveillance tests within the last surveillance interval. The requirement to complete this action before beginning a unit shutdown is to ensure that an alternate feedwater train is available before putting the affected unit through a transient. If no alternate feedwater trains are available, the affected unit is to stay at the same condition until an auxiliary feedwater train is returned to service, and then invoke ACTION statement 1 for the other train. If both standby feedwater pumps are made available before one auxiliary feedwater train is returned to an OPERABLE status, then the affected unit(s) shall be placed in at least HOT STANDBY within 6 hours and HOT SHUTDOWN within the following 6 hours.

ACTION statement 3 describes the actions to be taken when a single auxiliary feedwater pump is inoperable. The requirement to verify that two independent auxiliary feedwater trains are OPERABLE is to be accomplished by verifying that the requirements for Table 3.7-3 have been successfully met for each train within the last surveillance interval. The provisions of Specification 3.0.4 are not applicable to the third auxiliary feedwater pump provided it has not been inoperable for longer than 30 days. This means that a unit(s) can change OPERATIONAL MODES during a unit(s) heatup with a single auxiliary feedwater pump inoperable as long as the requirements of ACTION statement 3 are satisfied.

The monthly testing of the auxiliary feedwater pumps will verify their operability. Proper functioning of the turbine admission valve and the operation of the pumps will demonstrate the integrity of the system. Verification

INSERT TO TS BASES 3/4.7.1.1

Table 3.7-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

AUXILIARY FEEDWATER SYSTEM (Continued)

of correct operation will be made both from instrumentation within the control room and direct visual observation of the pumps.

indicated

3/4.7.1.3 CONDENSATE STORAGE TANK

indicated volume 210,000

There are two (2) seismically designed 250,000 gallons condensate storage tanks. A minimum of ~~185,000~~ gallons is maintained for each unit in MODES 1, 2 or 3. The OPERABILITY of the condensate storage tank with the minimum ~~water~~ volume ensures that sufficient water is available to maintain the Reactor Coolant System at HOT STANDBY conditions for approximately 23 hours or maintain the Reactor Coolant System at HOT STANDBY conditions for 15 hours and then cool down the Reactor Coolant System to below 350°F at which point the Residual Heat Removal System may be placed in operation.

INSERT ATTACHMENT HERE

3/4.7.1.4 SPECIFIC ACTIVITY

The limit on secondary coolant specific activity is based on a postulated release of secondary coolant equivalent to the contents of three steam generators to the atmosphere due to a net load rejection. The limiting dose for this case would result from radioactive iodine in the secondary coolant. One tenth of the iodine in the secondary coolant is assumed to reach the site boundary making allowance for plate-out and retention in water droplets. The inhalation thyroid dose at the site boundary is then;

DCF

$$\text{Dose (Rem)} = C * V * B * DFC * X/Q * 0.1$$

Where:

C	=	secondary coolant dose equivalent I-131 specific activity
	=	0.2 curies/m^3 (μCi/cc) or 0.1 Ci/m^3, each unit
V	=	equivalent secondary coolant volume released = 214 m^3
B	=	breathing rate = 3.47×10^{-4} m^3/sec.
X/Q	=	atmospheric dispersion parameter = 1.54×10^{-4} sec/m^3
0.1	=	equivalent fraction of activity released
DCF	=	dose conversion factor, Rem/Ci

The resultant thyroid dose is less than 1.5 Rem.

3/4.7.1.5 MAIN STEAM LINE ISOLATION VALVES

The OPERABILITY of the main steam line isolation valves ensures that no more than one steam generator will blow down in the event of a steam line rupture. This restriction is required to: (1) minimize the positive reactivity effects of the Reactor Coolant System cooldown associated with the blowdown, and (2) limit the pressure rise within containment in the event the steam line rupture occurs within containment. The OPERABILITY of the main steam isolation valves within the closure times of the Surveillance Requirements are consistent with the assumptions used in the safety analyses. The 24-hour action time provides a reasonable amount of time to troubleshoot and repair the backup air and/or nitrogen system.

INSERT TO TS BASES 3/4.7.1.3

The minimum indicated volume includes an allowance for instrument indication uncertainties and for water deemed unusable because of vortex formation and the configuration of the discharge line.

3/4.7.1.6 STANDBY STEAM GENERATOR FEEDWATER SYSTEM

The purpose of this specification and the supporting surveillance requirements is to assure operability of the non-safety grade Standby Steam Generator Feedwater System. The Standby Steam Generator Feedwater System consists of commercial grade components designed and constructed to industry and FPL standards of this class of equipment located in the outdoor plant environment typical of FPL facilities system wide. The system is expected to perform with high reliability, i.e., comparable to that typically achieved with this class of equipment. FPL intends to maintain the system in good operating condition with regard to appearance, structures, supports, component maintenance, calibrations, etc.

The function of the Standby Steam Generator Feedwater System for OPERABILITY determinations is that it can be used as a backup to the Auxiliary Feedwater (AFW) System in the event the AFW System does not function properly. The system would be manually started, aligned and controlled by the operator when needed.

The A pump is electric-driven and is powered from the non-safety related C bus. In the event of a coincident loss of offsite power, the B pump is diesel driven and can be started and operated independent of the availability of on-site or offsite power.

65,000

A supply of ~~60,000~~ gallons from the Demineralized Water Storage Tank for the Standby Steam Generator Feedwater Pumps is sufficient water to remove decay heat from the reactor for six (6) hours for a single unit or two (2) hours for two units. This was the basis used for requiring ~~60,000~~ gallons of water in the non-safety grade Demineralized Water Storage Tank and is judged to provide sufficient time for restoring the AFW System or establishing make-up to the Demineralized Water Storage Tank.

INSERT ATTACHMENT HERE

The Standby Steam Generator Feedwater Pumps are not designed to NRC requirements applicable to Auxiliary Feedwater Systems and are not required to satisfy design basis events requirements. These pumps may be out of service for up to 24 hours before initiating formal notification because of the extremely low probability of a demand for their operation.

The guidelines for NRC notification in case of both pumps being out of service for longer than 24 hours are provided in applicable plant procedures, as a voluntary 4-hour notification.

Adequate demineralized water for the Standby Steam Generator Feedwater system will be verified once per 24 hours. The Demineralized Water Storage Tank provides a source of water to several systems and therefore, requires daily verification.

The Standby Steam Generator Feedwater Pumps will be verified OPERABLE monthly on a STAGGERED TEST BASIS by starting and operating them in the recirculation mode. Also, during each unit's refueling outage, each Standby Steam Generator Feedwater Pump will be started and aligned to provide flow to the nuclear unit's steam generators.

INSERT TO TS BASES 3/4.7.1.6

The minimum indicated volume (135,000 gallons) consists of an allowance for level indication instrument uncertainties (approximately 15,000 gallons); for water deemed unusable because of tank discharge line location and vortex formation (approximately 50,300 gallons); and the minimum usable volume (65,000 gallons). The minimum indicated volume corresponds to a water level of 8.5 feet in the Demineralized Water Storage Tank.

CORE OPERATING LIMITS REPORT

FOR

FLORIDA POWER AND LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4

CORE OPERATING LIMITS REPORT

1.0 INTRODUCTION

The Core Operating Limits Report (COLR) for Turkey Point Unit 3 Cycle 15 has been prepared in accordance with the requirements of Technical Specifications 6.9.1.7.

The Technical Specifications affected by this report are:

3.2.1 Axial Flux Difference (AFD)
3.1.3.6 Control Rod Insertion Limits
3/4.2.2 Heat Flux Hot Channel Factor - $F_Q(Z)$
3/4.2.3 Nuclear Enthalpy Rise Hot Channel Factor

2.0 OPERATING LIMITS

The AFD, $F_Q(Z)$, $F_{\Delta H}$, K(Z) and Rod Bank Insertion Limits have been developed using the NRC approved methodology specified in Technical Specification 6.9.1.7. These limits are provided in the following subsections:

2.1 Axial Flux Difference (TS 3.2.1)

The Axial Flux Difference (AFD) limits are provided in Figure 1.

2.2 Control Rod Insertion Limits (TS 3.1.3.6)

The control rod banks shall be limited in physical insertion as shown in Figure 2.

2.3 Heat Flux Hot Channel Factor $F_Q(Z)$ (TS 3/4.2.2)

- o $[F_Q]^L$ = 2.32
- o K(Z) is provided in Figure 3.

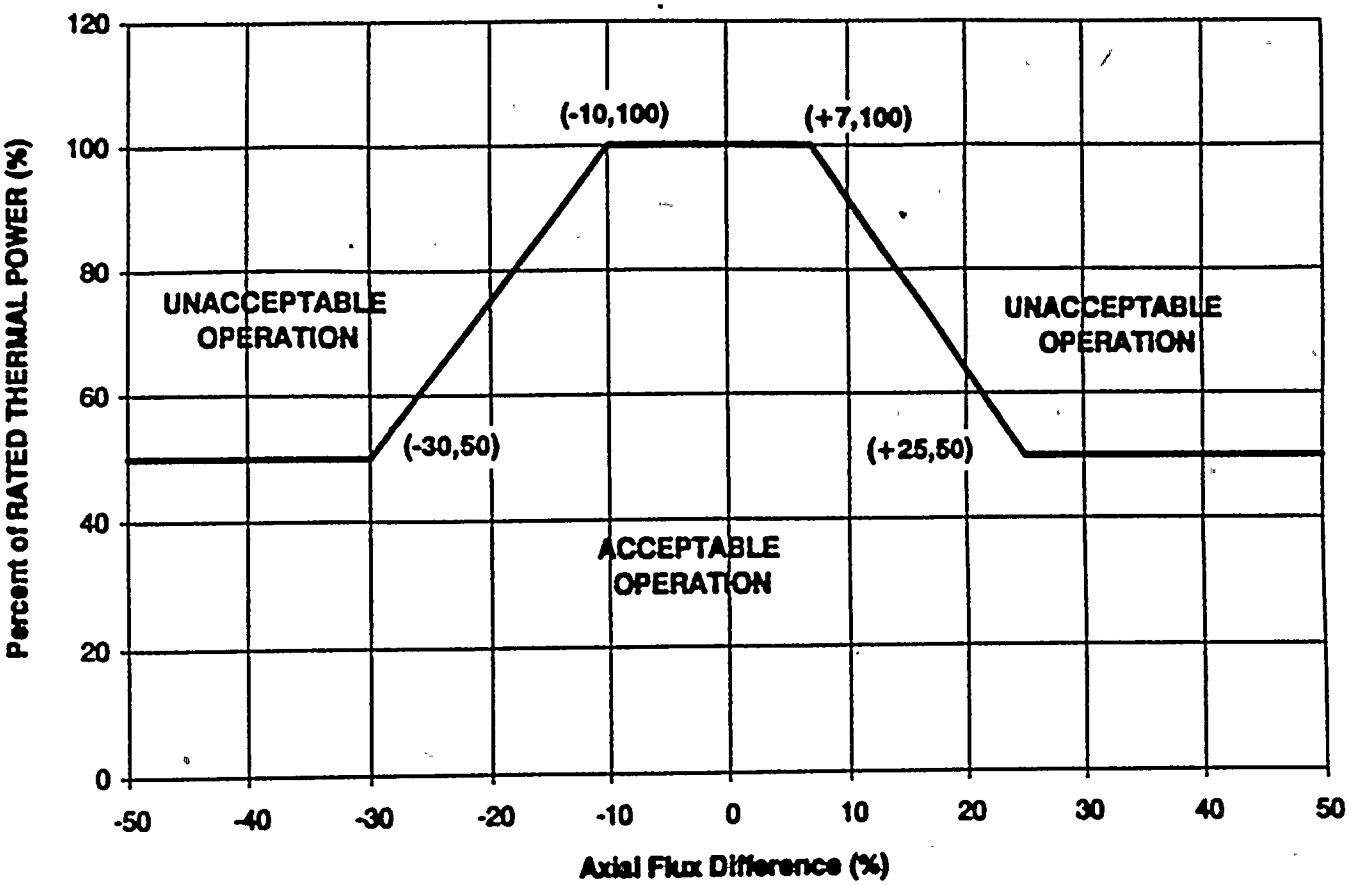
2.4 Nuclear Enthalpy Rise Hot Channel Factor

- o $F_{\Delta H}^{RTP}$ = 1.62
- o $PF_{\Delta H}$ = 0.3

L-95-245
Attachment 3

FIGURE 1

AXIAL FLUX DIFFERENCE AS A FUNCTION OF RATED THERMAL POWER (TURKEY POINT UNIT 3 CYCLE 15)



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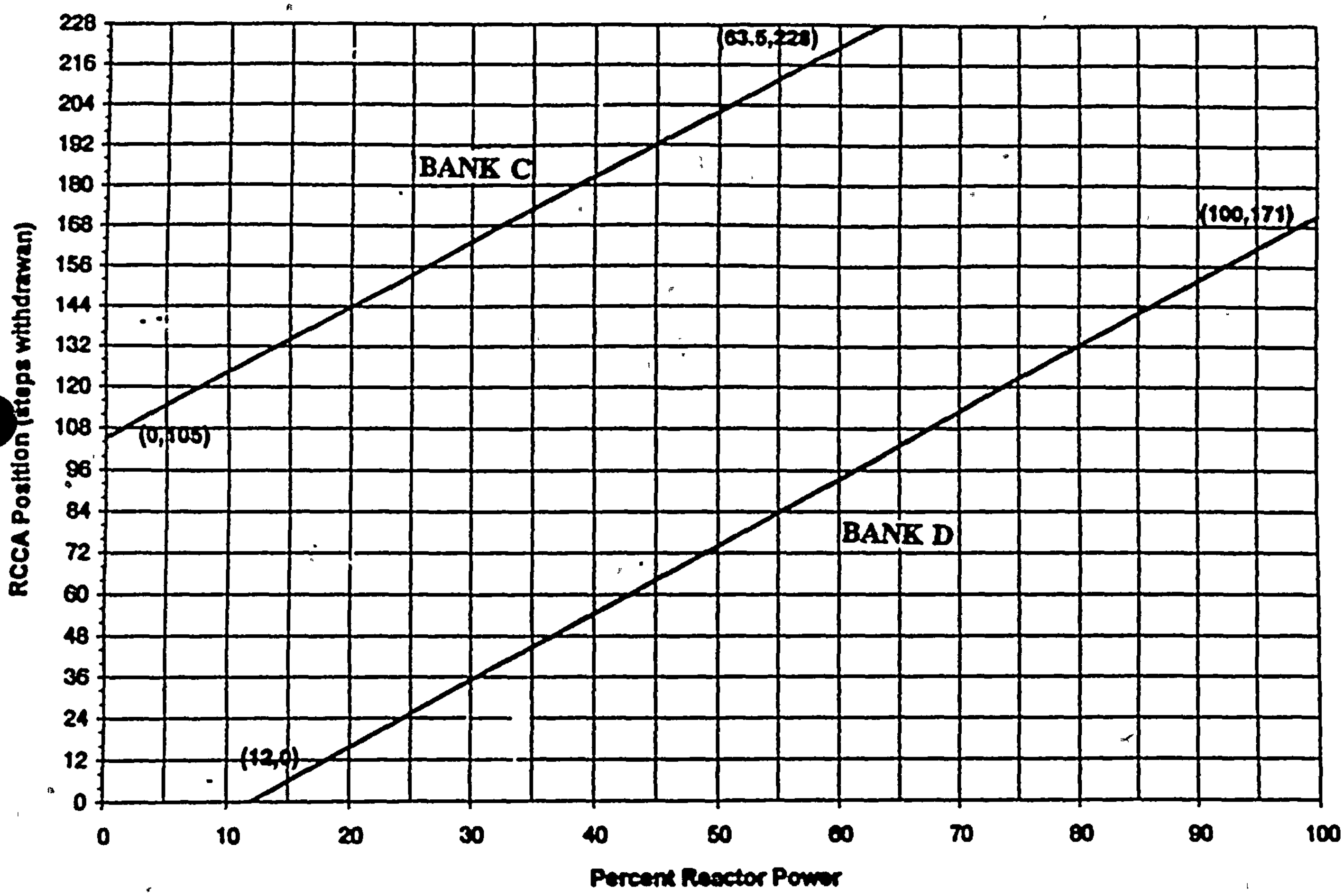
Attachment 3

FIGURE 2

ROD BANK INSERTION LIMITS VS. THERMAL POWER
ARO = 228 STEPS WITHDRAWN, OVERLAP = 100 STEPS
(TURKEY POINT UNIT 3 CYCLE 15)



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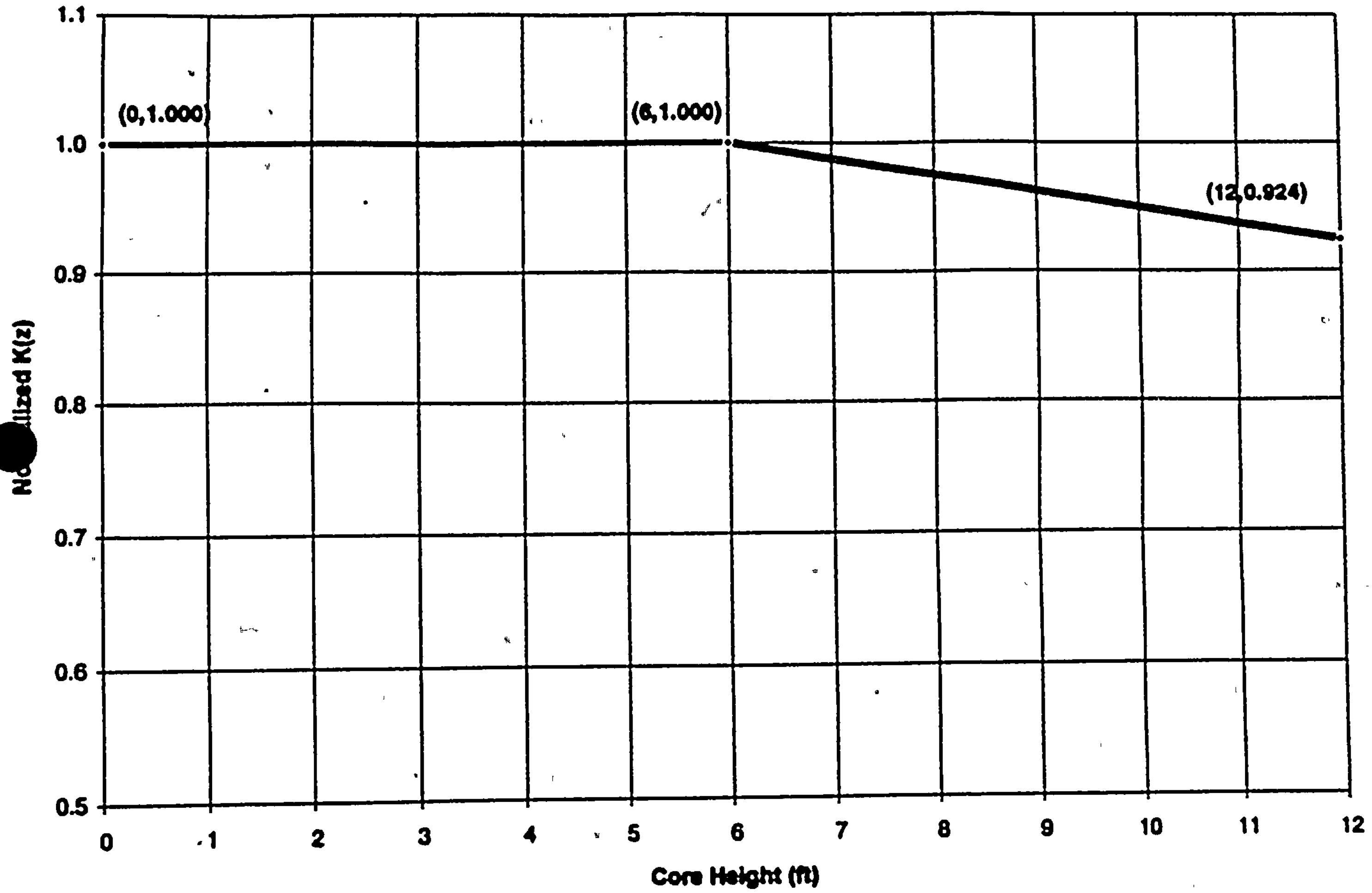
Attachment 3

FIGURE 3

K(Z) NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT
(TURKEY POINT UNIT 3 CYCLE 15)



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Attachment 3

CORE OPERATING LIMITS REPORT

1.0 INTRODUCTION

The Core Operating Limits Report (COLR) for Turkey Point Unit 4 Cycle 15 has been prepared in accordance with the requirements of Technical Specifications 6.9.1.7.

The Technical Specifications affected by this report are:

3.2.1 Axial Flux Difference (AFD)
3.1.3.6 Control Rod Insertion Limits
3/4.2.2 Heat Flux Hot Channel Factor - $F_Q(Z)$
3/4.2.3 Nuclear Enthalpy Rise Hot Channel Factor

2.0 OPERATING LIMITS

The AFD, $F_Q(Z)$, $F_{\Delta H}$, K(Z) and Rod Bank Insertion Limits have been developed using the NRC approved methodology specified in Technical Specification 6.9.1.7. These limits are provided in the following subsections:

2.1 Axial Flux Difference (TS 3.2.1)

The Axial Flux Difference (AFD) limits are provided in Figure 1.

2.2 Control Rod Insertion Limits (TS 3.1.3.6)

The control rod banks shall be limited in physical insertion as shown in Figure 2.

2.3 Heat Flux Hot Channel Factor $F_Q(Z)$ (TS 3/4.2.2)

- o $[F_Q]^L$ = 2.32
- o K(Z) is provided in Figure 3.

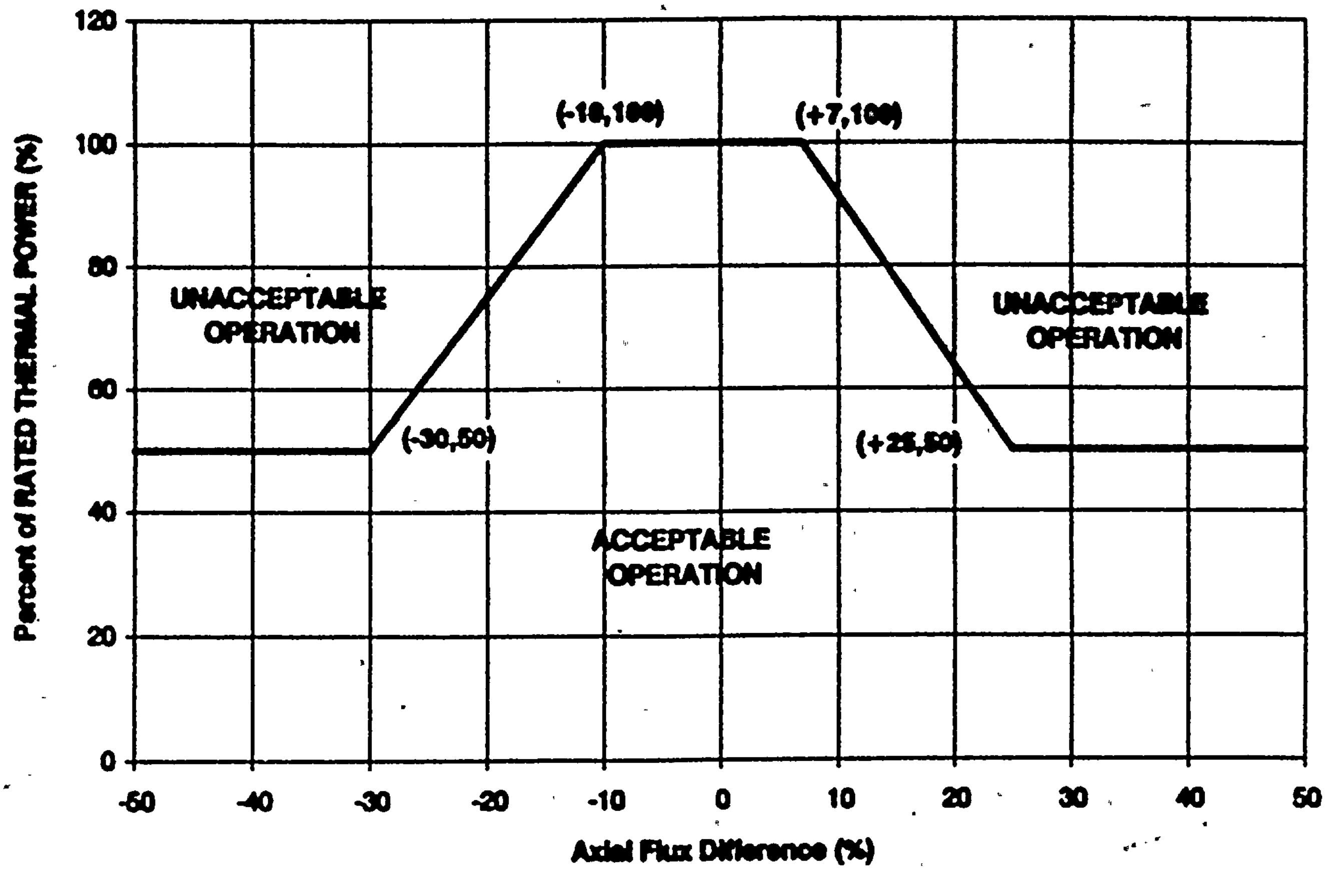
2.4 Nuclear Enthalpy Rise Hot Channel Factor

- o $F_{\Delta H}^{RTP}$ = 1.62
- o $PF_{\Delta H}$ = 0.3

L-95-245
Attachment 3

FIGURE 1

AXIAL FLUX DIFFERENCE AS A FUNCTION OF RATED THERMAL POWER
(TURKEY POINT UNIT 4 CYCLE 15)



L-95-245
Attachment 3

FIGURE 2

ROD BANK INSERTION LIMITS VS. THERMAL POWER
ARO = 228 STEPS WITHDRAWN, OVERLAP = 100 STEPS
(TURKEY POINT UNIT 4 CYCLE 15)

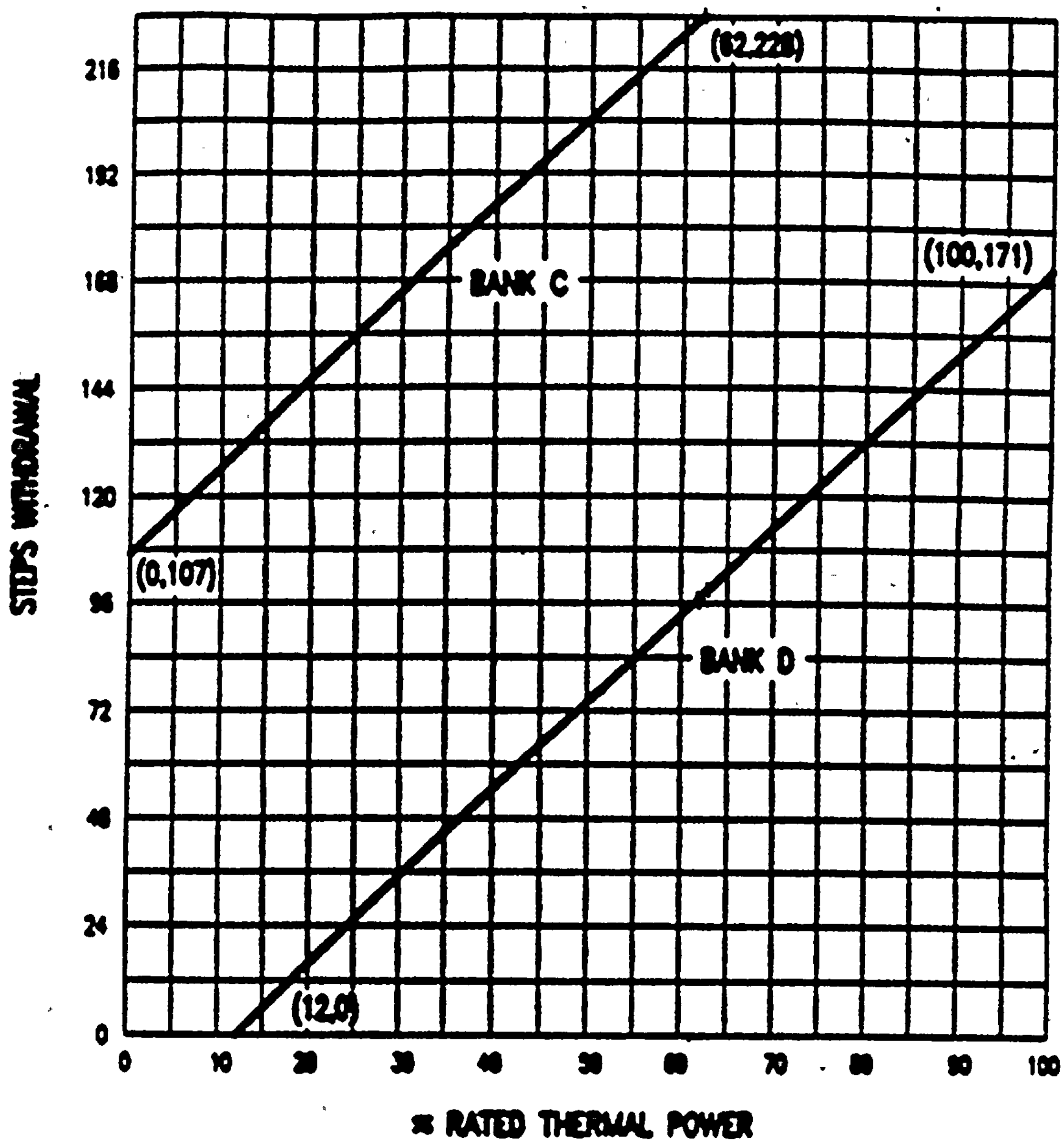
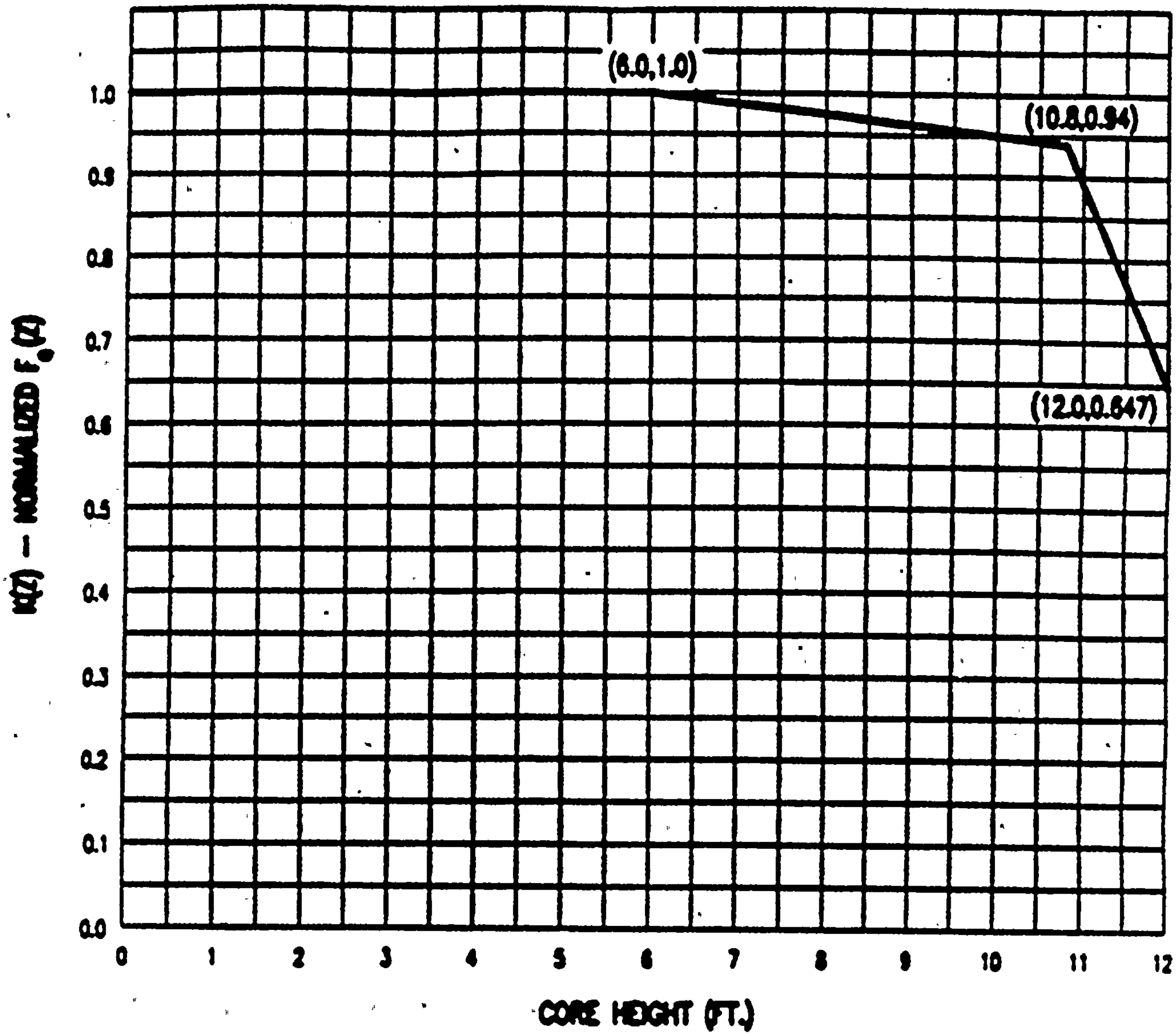


FIGURE 3

K(Z) NORMALIZED $F_Q(Z)$ AS A FUNCTION OF CORE HEIGHT
(TURKEY POINT UNIT 4 CYCLE 15)



ENCLOSURE 1

WCAP-14276

FLORIDA POWER & LIGHT COMPANY
TURKEY POINT UNITS 3 AND 4
UPRATING LICENSING REPORT

ENCLOSURE 2

PROPRIETARY WCAP 13719, REVISION 2

WESTINGHOUSE REVISED THERMAL DESIGN PROCEDURE
INSTRUMENTS UNCERTAINTY FOR
TURKEY POINT UNITS 3 AND 4

and

NON-PROPRIETARY WCAP 13718, REVISION 2

WESTINGHOUSE REVISED THERMAL DESIGN PROCEDURE
INSTRUMENTS UNCERTAINTY FOR
TURKEY POINT UNITS 3 AND 4

ENCLOSURE 3

WESTINGHOUSE AUTHORIZATION LETTER CAW-95-890

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