Turkey Point Units 3 and 4

LICENSE AMENDMENT REQUEST FOR EXTENDED POWER UPRATE

ATTACHMENT 2

Renewed Facility Operating License and
Technical Specification Proposed Change Markups

This coversheet plus 91 pages

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-250

TURKEY POINT NUCLEAR GENERATING UNIT NO. 3

RENEWED FACILITY OPERATING LICENSE NO. DPR-31

The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-31 issued on July 19, 1972, has now found that:

a. The application to renew License No. DPR-31 filed by Florida Power and Light Company, 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 and all required notifications to other agencies or bodies have been duly made;

b. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the Turkey Point Unit 3 plant, and that any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations;

c. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

d. There is reasonable assurance (i) that the facility can be operated at steady state power levels up to ~~2300~~ 2644 megawatts thermal in accordance with this renewed operating license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. Florida Power and Light Company is technically and financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

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 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 renewed operating 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, Sections 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 applicant is authorized to operate the facility at reactor core power levels not in excess of ~~2300~~ 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 241 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than July 19, 2012.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

FLORIDA POWER & LIGHT COMPANY

DOCKET NO. 50-251

TURKEY POINT NUCLEAR GENERATING UNIT NO. 4

RENEWED FACILITY OPERATING LICENSE NO. DPR-41

The U.S. Nuclear Regulatory Commission (the Commission) having previously made the findings set forth in License No. DPR-41 issued on April 10, 1973, has now found that:

a. The application to renew License No. DPR-41 filed by Florida Power and Light Company, 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 and all required notifications to other agencies or bodies have been duly made;

b. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the Turkey Point Unit 4 plant, and that any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations;

c. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission;

d. There is reasonable assurance (i) that the facility can be operated at steady state power levels up to ~~2300~~ 2644 megawatts thermal in accordance with this renewed operating license without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. Florida Power and Light Company is technically and financially qualified to engage in the activities authorized by this renewed operating license in accordance with the rules and regulations of the Commission;

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 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 renewed operating 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, Sections 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 applicant is authorized to operate the facility at reactor core power levels not in excess of ~~2300~~ 2644 megawatts (thermal).

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 237 are hereby incorporated into this renewed license. The Environmental Protection Plan contained in Appendix B is hereby incorporated into this renewed license. The licensee shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

C. Final Safety Analysis Report

The licensee's Final Safety Analysis Report supplement submitted pursuant to 10 CFR 54.21(d), as revised on November 1, 2001, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than April 10, 2013.

The Final Safety Analysis Report supplement as revised on November 1, 2001, described above, shall be included in the next scheduled update to the Final Safety Analysis Report required by 10 CFR 50.71(e)(4), following the issuance of this renewed license. Until that update is complete, the licensee may make changes to the programs described in such supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

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QUADRANT POWER TILT RATIO

1.22 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

1.23 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~2300~~ 2644 MWt.

REPORTABLE EVENT

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

SHUTDOWN MARGIN

1.25 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

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

SOLIDIFICATION

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

SOURCE CHECK

1.28 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

1.29 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.

See New Figure 2.1-1 below

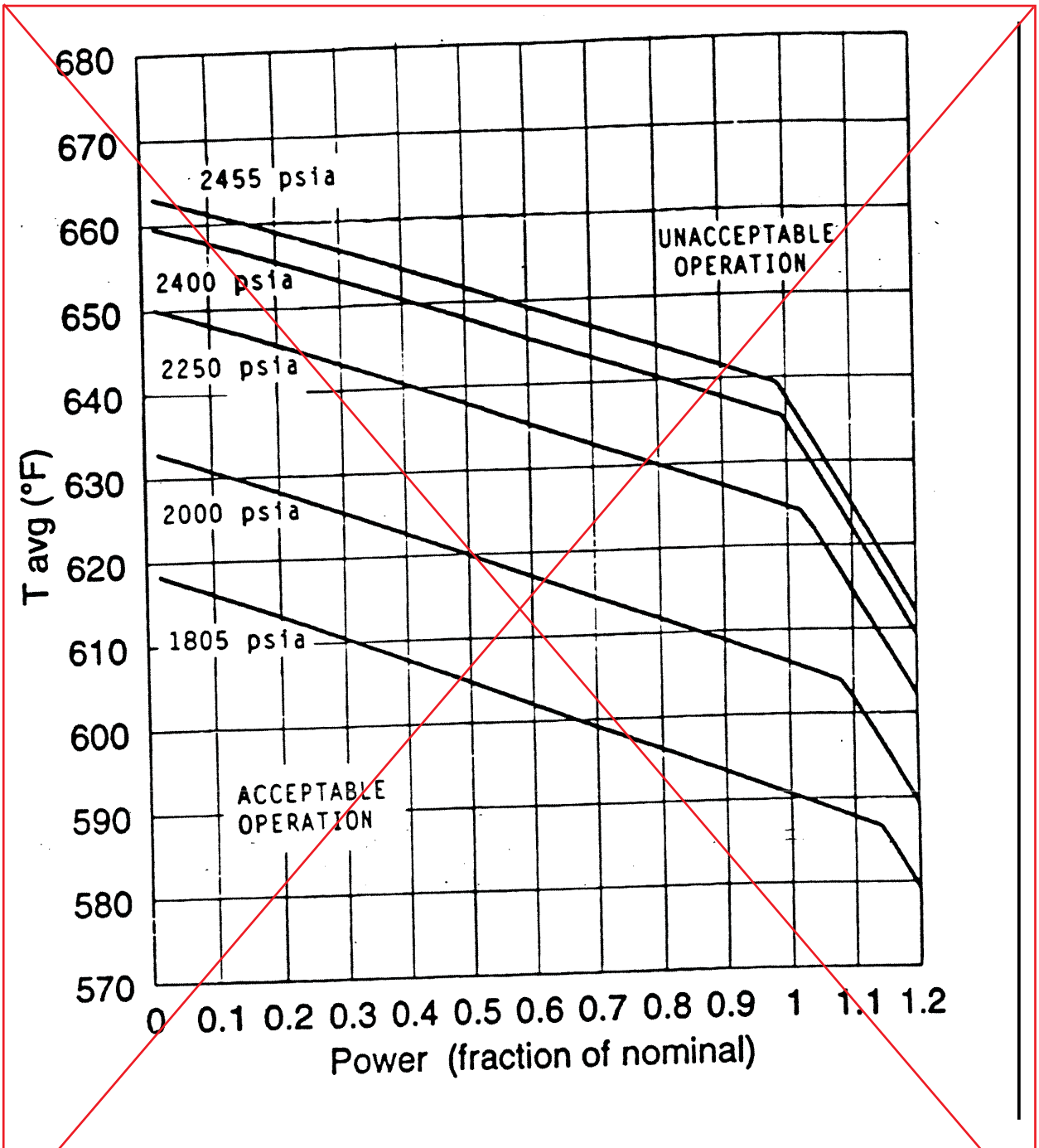

Figure 2.1-1 Reactor Core Safety Limit – Three Loops in Operation

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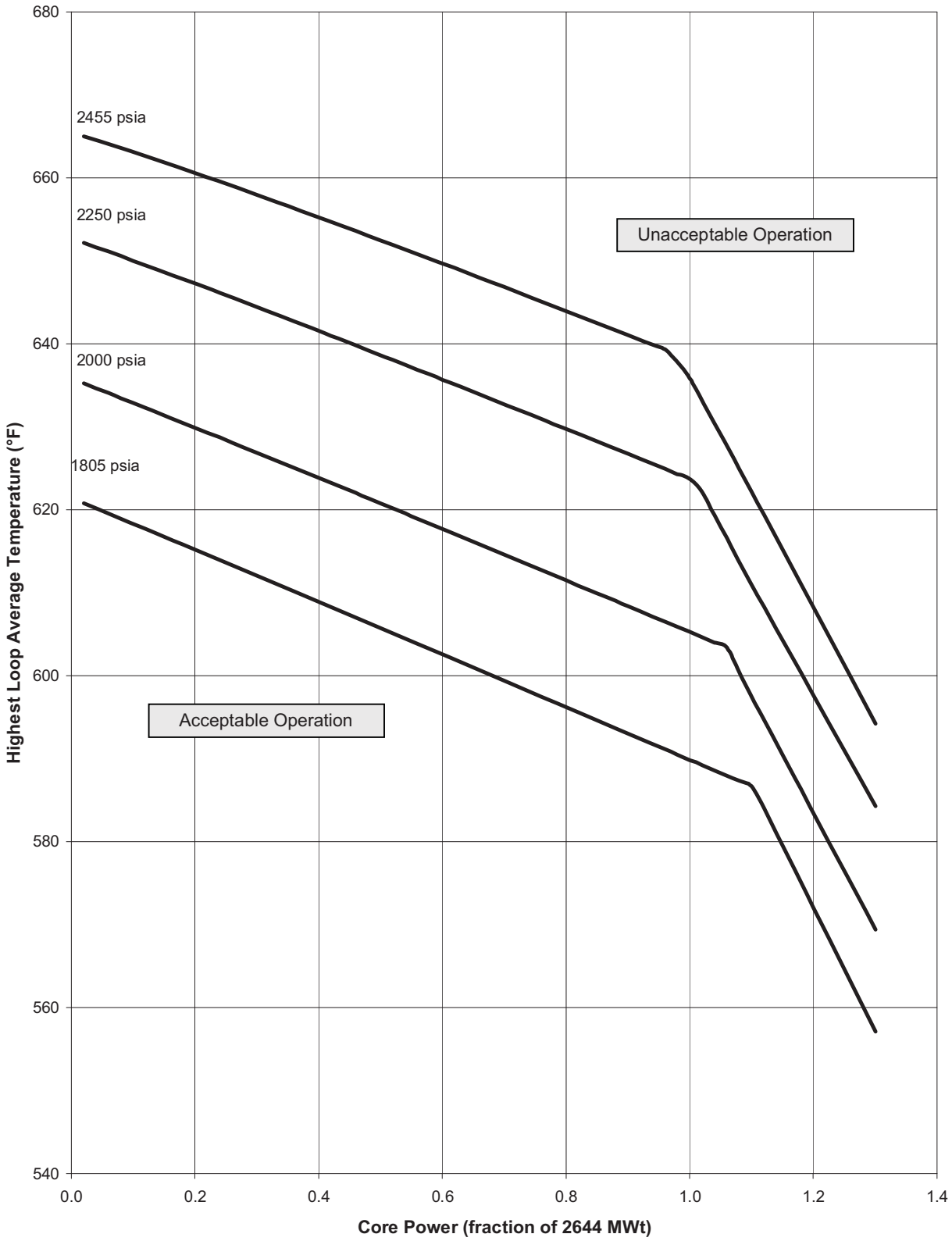


Figure 2.1-1 Reactor Core Safety Limit Three Loops in Operation

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	109.6 ≤ ~~112.0~~% of RTP**	(a),(b) ≤ 109.0% 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 (a), (b) for K_1
6. Overpower ΔT	See Note 4	See Note 3 (a), (b) for K_4
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	89.6 ≥ ~~88.8~~% of loop design flow*	(a),(b) ≥ 90% of loop design flow*
11. Steam Generator Water Level Low-Low	15.5 ≥ ~~8.15~~% of narrow range instrument span	16 ≥ ~~10~~% of narrow range instrument span

86,900

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
12. Steam/Feedwater Flow Mismatch Coincident with	20.7 Feed Flow ≤ ~~23.9~~% below rated Steam Flow	Feed Flow ≤ 20% below rated Steam Flow (a),(b)
Steam Generator Water Level-Low	15.5 ≥ ~~8.15~~% of narrow range instrument span	16 ≥ ~~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 (Emergency Trip Header)	≥ ~~42~~ psig (950)	≥ ~~45~~ psig (1000) (a), (b)
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} amps

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in WCAP-17070-P.

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

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT

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

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~~ 2s

ΔT_0 = Indicated ΔT at RATED THERMAL POWER

K_1 = ~~1.24;~~ 1.31 (a), (b)

K_2 = ~~0.017/°F;~~ 0.023/°F

$\frac{1+\tau_4 S}{1+\tau_5 S}$ = 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 = 3s;

T = Average temperature, °F;

$\frac{1}{1+\tau_6 S}$ = Lag compensator on measured T_{avg}; τ_6 = ~~0s~~ 2s

T′ ≤ 583 ~~577.2~~ °F (~~Nominal~~ Indicated Loop T_{avg} at RATED THERMAL POWER);

K_3 = ~~0.001/psig~~; 0.00116/ psi

P = Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

P' ≥ 2235 psig (Nominal RCS operating pressure);

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

And $f_1(\Delta I)$ 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 – ~~50%~~ 18% and + ~~2%~~ 7%, $f_1(\Delta I) = 0$, 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 – ~~50%~~ 18%, the ΔT Trip Setpoint shall be automatically reduced by ~~0.0%~~ 3.51% of its value at RATED THERMAL POWER; and

(3) For each percent that the magnitude of $q_t - q_b$ exceeds + ~~2%, that~~ 7%, the ΔT Trip Setpoint shall be automatically reduced by ~~2.19%~~ 2.37% of its value at RATED THERMAL POWER.

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

The Overtemperature ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel, 0.2% ΔT span for the Pressurizer Pressure channel, and 0.4% ΔT span for the f(ΔI) channel. No separate Allowable Value is provided for Tavg because this function is part of the ΔT value.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in WCAP-17070-P.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT

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

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	≤	1.10, (a), (b)
K_5	≥	~~0.02~~ 0.0/°F for increasing average temperature and ~~0~~ 0/°F for decreasing average temperature,
$\frac{\tau_7 s}{1+\tau_7 S}$	=	The function generated by the lead-lag compensator for T_{avg} dynamic compensation;
τ_7	=	Time constants utilized in the lead-lag compensator for T_{avg}, $\tau_7 \geq$ ~~10 s,~~ 0 s,
$\frac{1}{1+\tau_6 S}$	=	As defined in Note 1,

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: (Continued)

K_6	=	0.0016/°F for T > T"
	=	0 for T ≤ T",
T	=	583 As defined in Note 1, Indicated Loop
T"	≤	~~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 0.96% of instrument span.~~

The Overpower ΔT function Allowable Value shall not exceed the nominal trip setpoint by more than 0.5% ΔT span for the ΔT channel. No separate Allowable Value is provided for Tavg because this function is part of the ΔT value.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in WCAP-17070-P.

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be:

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

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

d. Less negative than ~~-3.5~~ -4.1 $\times 10^{-4}$ $\Delta k/k/°F$ for the all rods withdrawn, end of cycle life (EOL), RATED THERMAL POWER condition.

APPLICABILITY: Specification 3.1.1.3a, b and c. - MODES 1 and 2* only**.
Specification 3.1.1.3d. - MODES 1, 2, and 3 only**.

ACTION:

a. With the MTC more positive than the limit of Specification 3.1.1.3a, 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 3.1.1.3a, 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 3.1.3.6;

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 6.9.2, 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 positive MTC to within its limit for the all rods withdrawn condition.

* With K_{eff} greater than or equal to 1.
** See Special Test Exceptions Specification 3.10.3.

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

b. With the MTC more negative than the limit of Specification 3.1.1.3d. above, be in HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.1.1.3 The MTC shall be determined to be within its limits during each fuel cycle as follows:

a. The MTC shall be measured and compared to the BOL limit of Specification 3.1.1.3a., above, prior to initial operation above 5% of RATED THERMAL POWER, after each fuel loading; and

b. The MTC shall be measured at any THERMAL POWER and compared to ~~-3.0~~ -3.5 x 10^{-4} $\Delta k/k/°F$ (all rods withdrawn, RATED THERMAL POWER condition) within 7 EFPD after reaching an equilibrium boron concentration of 300 ppm. In the event this comparison indicates the MTC is more negative than ~~-3.0~~ -3.5 x 10^{-4} $\Delta k/k/°F$, the MTC shall be remeasured, and compared to the EOL MTC limit of Specification 3.1.1.3d., at least once per 14 EFPD during the remainder of the fuel cycle.

c. Perform design calculation to verify conformance to Specifications 3.1.1.3b and c.

3/4.1.2 BORATION SYSTEMS

FLOW PATH - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.1 As a minimum, one of the following boron injection flow paths shall be OPERABLE and capable of being powered from an OPERABLE emergency power source:

a. A flow path from the boric acid storage tanks via a boric acid transfer pump and a charging pump to the Reactor Coolant System if the boric acid storage tank in Specification 3.1.2.4a. is OPERABLE, or

b. The flow path from the refueling water storage tank via a charging pump to the Reactor Coolant System if the refueling water storage tank in Specification 3.1.2.4b. is OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

With none of the above flow paths OPERABLE or capable of being powered from an OPERABLE emergency power source, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEILLANCE REQUIREMENTS

4.1.2.1 At least one of the above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the rooms containing flow path components is greater than or equal to ~~55°F~~ 62°F when a flow path from the boric acid tanks is used, and

b. At least once per 31 days by 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.

SURVEI LLANCE REQUIREMENTS

4.1.2.2 The above required flow paths shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that the temperature of the rooms containing flow path components is greater than or equal to ~~55°F~~ 62°F when a flow path from the boric acid tanks is used;

b. At least once per 31 days by 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;

c. At least once per 18 months by verifying that the flow path required by Specification 3.1.2.2a. and c. delivers at least 16 gpm to the RCS.

BORATED WATER SOURCE - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.1.2.4 As a minimum, one of the following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum indicated borated water volume of 2,900 gallons per unit,

2) A boron concentration between 3.0 wt% (5245 ppm) and ~~3.5~~ 4.0 wt.% (~~6119~~ 6993 ppm), and

3) A minimum boric acid tanks room temperature of ~~55°F~~ 62°F.

b. The refueling water storage tank (RWST) with:

1) A minimum indicated borated water volume of 20,000 gallons,

2) A ~~minimum~~ boron concentration ~~of 1950 ppm~~ between 2400 ppm and 2600 ppm, and

3) A minimum solution temperature of 39°F.

APPLICABILITY: MODES 5 and 6.

ACTION:

With no borated water source OPERABLE, suspend all operations involving CORE ALTERATIONS or positive reactivity changes.

SURVEI LLANCE REQUI REMENTS

4.1.2.4 The above required borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration of the water,

2) Verifying the indicated borated water volume, and

3) Verifying that the temperature of the boric acid tanks room is greater than or equal to ~~55°F~~ 62°F, when it is the source of borated water.

LIMITING CONDITION FOR OPERATION

3.1.2.5 The following borated water sources shall be OPERABLE:

a. A Boric Acid Storage System with:

1) A minimum indicated borated water volume in accordance with Figure 3.1-2,

2) A boron concentration in accordance with Figure 3.1-2, and

3) A minimum boric acid tanks room temperature of ~~55°F~~ 62°F.

b. The refueling water storage tank (RWST) with:

1) A minimum indicated borated water volume of 320,000 gallons,

2) A ~~minimum~~ boron concentration ~~of 1950 ppm~~ between 2400 ppm and 2600 ppm.

3) A minimum solution temperature of 39°F, and

4) A maximum solution temperature of 100°F.

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

ACTION:

a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours* and borated to a SHUTDOWN MARGIN equivalent to at least 1% Δk/k at 200°F; restore the Boric Acid Storage System to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.

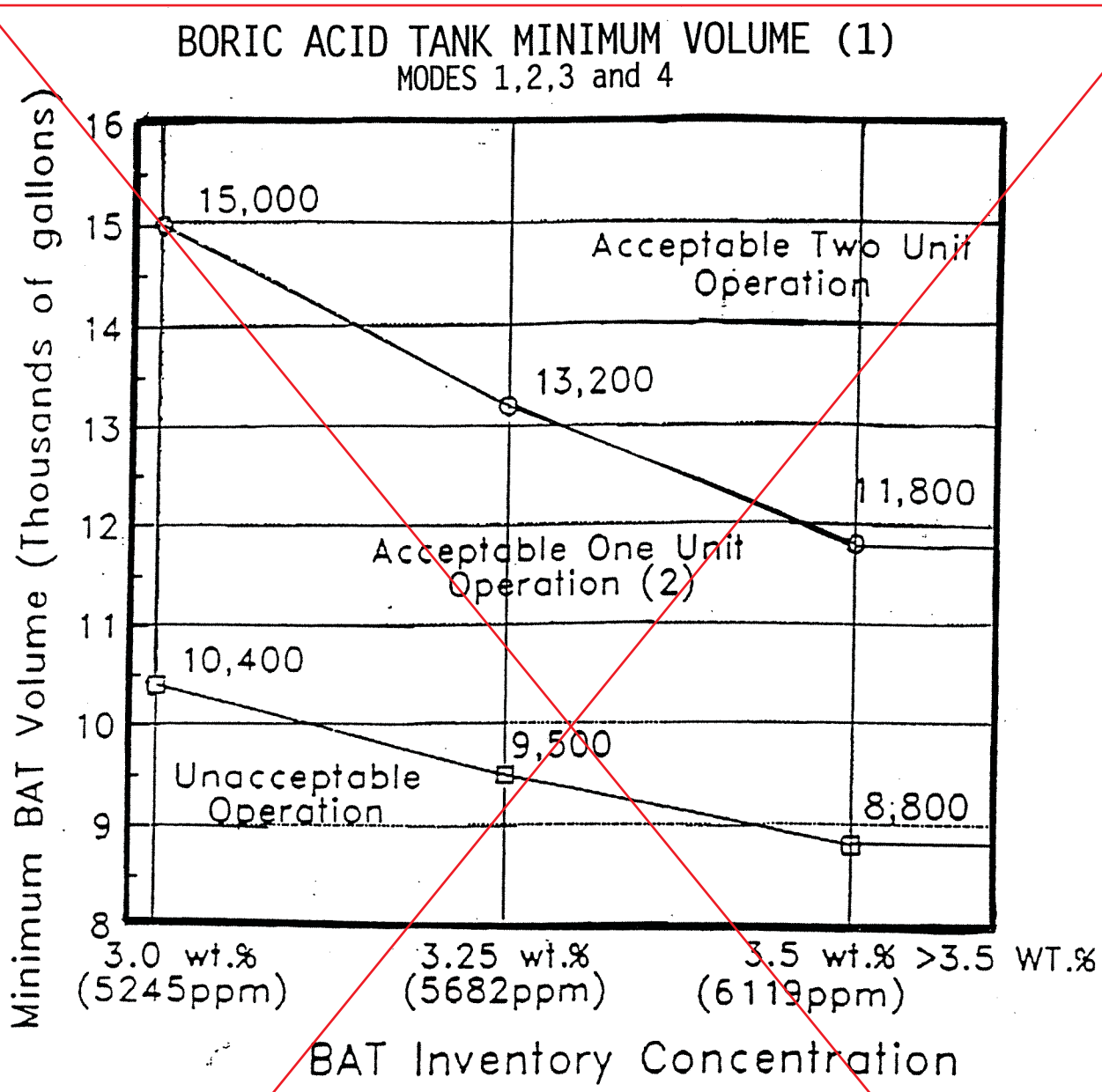
b. With the RWST inoperable, restore the tank to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With the boric acid tank inventory concentration greater than ~~3.5 wt%~~ 4.0 wt%, verify that the boric acid solution temperature for boration sources and flow paths is greater than the solubility limit for the concentration.

* If this action applies to both units simultaneously, be in at least HOT STANDBY within the next twelve hours.

Replace with new Figure 3.1-2 on next page

~~FIGURE 3.1-2~~



Notes:

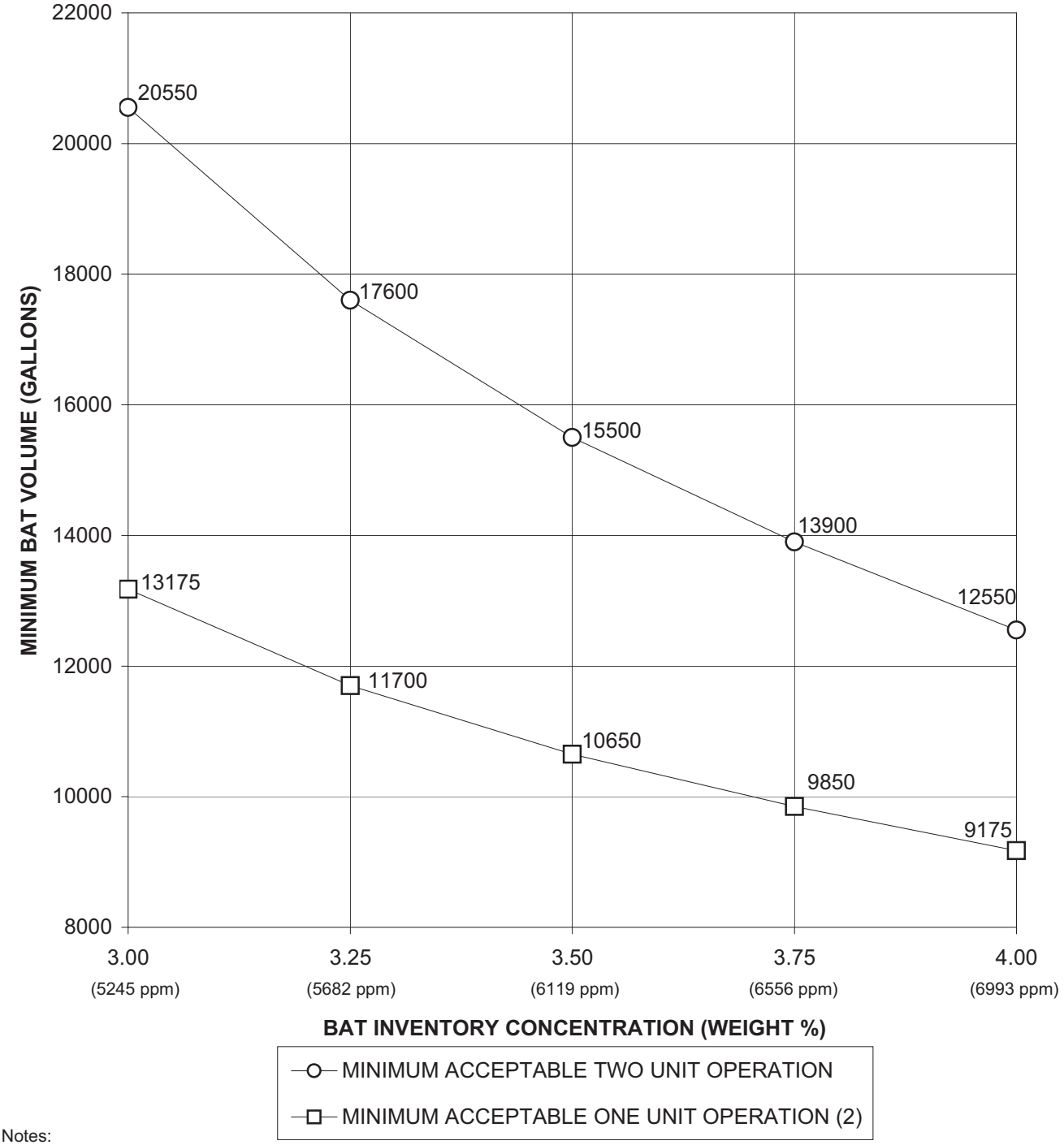
(1) Combined volume of all available boric acid tanks assuming RWST boron concentration greater than or equal to 1950 ppm.

(2) Includes 2900 gallons for shutdown unit.

FIGURE 3.1-2

BORIC ACID TANK MINIMUM VOLUME (1)

MODES 1, 2, 3 AND 4



Notes:

(1) Combined volume of all available boric acid tanks assuming RWST boron concentration between 2400 ppm and 2600 ppm.
(2) Includes 2900 gallons for the shutdown unit.

SURVEILLANCE REQUIREMENTS

4.1.2.5 Each borated water source shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the boron concentration in the water,

2) Verifying the indicated borated water volume of the water source, and

62°F

3) Verifying that the temperature of the boric acid tanks room is greater than or equal to ~~55°F~~, when it is the source of borated water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for 23 consecutive hours, and

2) At least once per 24 hours while the outside temperature exceeds its limits.

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

585

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

2204

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

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

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

270,000

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 then 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

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.

* 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.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Steam Generator Water Level--Low-Low	3/stm. gen.	2/stm. gen.	2/stm. gen.	1, 2	6
12. Steam Generator Water Level--Low Coincident With Steam/Feedwater Flow Mismatch	2 stm. gen. level and 2 stm./feed-water flow mismatch in each stm. gen.	1 stm. gen. level coin-cident with 1 stm./feed-water flow mismatch in same stm. gen.	1 stm. gen. level and 2 stm./feed-water flow mismatch in same stm. gen. or 2 stm. gen. level and 1 stm./feedwater flow mismatch in same stm. gen.	1, 2	6
13. Undervoltage--4.16 KV Busses A and B (Above P-7)	2/bus	1/bus on both busses	2/bus	1	12
14. Underfrequency-Trip of Reactor Coolant Pump Breaker(s) Open (Above P-7)	2/bus	1 to trip RCPs***	2/bus	1	11
15. Turbine Trip (Above P-7)					
a. ~~Autostop Oil~~ Pressure	3	2	2	1	12
b. Turbine Stop Valve Closure	2	2	2	1	12

Emergency Trip Header

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
1. Manual Reactor Trip	N.A.	N.A.	N.A.	R(11)	N.A.	1, 2, 3*, 4*, 5*
2. Power Range, Neutron Flux						
a. High Setpoint	S	D(2, 4), M(3, 4), Q(4, 6), R(4)	Q	N.A.	N.A.	1, 2
b. Low Setpoint	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
3. Intermediate Range, Neutron Flux	S	R(4)	S/U(1)	N.A.	N.A.	1***, 2
4. Source Range, Neutron Flux	S	R(4)	S/U(1), Q(9)	N.A.	N.A.	2**,3, 4, 5
5. Overtemperature ΔT	S	R	Q	N.A.	N.A.	1, 2
6. Overpower ΔT	S	R	Q	N.A.	N.A.	1, 2
7. Pressurizer Pressure--Low	S	R	Q	N.A.	N.A.	1
8. Pressurizer Pressure--High	S	R	Q	N.A.	N.A.	1, 2
9. Pressurizer Water Level--High	S	R	Q	N.A.	N.A.	1
10. Reactor Coolant Flow--Low	S	R	Q	N.A.	N.A.	1
11. Steam Generator Water Level--Low-Low	S	R	Q	N.A.	N.A.	1, 2

(a),(b)

(a), (b)

(a), (b)*

(a), (b)**

(a), (b)*

(a), (b)**

(a),(b)

(a),(b)

* Notes (a) and (b) are applicable to K_1. See Table 2.1-1, Note 1.
** Notes (a) and (b) are applicable to K_4. See Table 2.1-1, Note 3.

Emergency Trip Header

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST	MODES FOR WHICH SURVEILLANCE IS REQUIRED
12. Steam Generator Water Level--Low Coincident with Steam/Feedwater Flow Mismatch	S	R (a),(b)	Q (a),(b)	N.A.	N.A.	1, 2
13. Undervoltage – 4.16 kV Busses A and B	N.A.	R	N.A.	N.A.	N.A.	1
14. Underfrequency – Trip of Reactor Coolant Pump Breakers(s) Open	N.A.	R	N.A.	N.A.	N.A.	1
15. Turbine Trip		(a), (b)				
a. ~~Autostop Oil~~ Pressure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
b. Turbine Stop Valve Closure	N.A.	R	N.A.	S/U(1, 10)	N.A.	1
16. Safety Injection Input from ESF	N.A.	N.A.	N.A.	R	N.A.	1, 2
17. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6	N.A.	R(4)	R	N.A.	N.A.	2**
b. Low Power Reactor Trips Block, P-7 (includes P-10 input and Turbine First Stage Pressure)	N.A.	R(4)	R	N.A.	N.A.	1
c. Power Range Neutron Flux, P-8	N.A.	R(4)	R	N.A.	N.A.	1

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

TABLE 4.3-1 (Continued)

TABLE NOTATIONS

* When the Reactor Trip System breakers are closed and the Control Rod Drive System is capable of rod withdrawal.

** Below P-6 (Intermediate Range Neutron Flux Interlock) Setpoint.

*** Below P-10 (Low Setpoint Power Range Neutron Flux Interlock) Setpoint.

(1) If not performed in previous 31 days.

(2) Comparison of calorimetric to excore power level indication above 15% of RATED THERMAL POWER (RTP). Adjust excore channel gains consistent with calorimetric power level if the absolute difference is greater than 2%. Below 70% RTP, downward adjustments of NIS excore channel gains to match a lower calorimetric power level are not required. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(3) Single point comparison of incore to excore AXIAL FLUX DIFFERENCE above 15% of RATED THERMAL POWER. Recalibrate if the absolute difference is greater than or equal to 3%. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(4) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(5) This table Notation number is not used.

(6) Incore-Excore Calibration, above 75% of RATED THERMAL POWER (RTP). If the quarterly surveillance requirement coincides with sustained operation between 30% and 75% of RTP, calibration shall be performed at this lower power level. The provisions of Specification 4.0.4 are not applicable for entry into MODE 2 or 1.

(7) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(8) DELETED

(9) Quarterly surveillance in MODES 3*, 4*, and 5* shall also include verification that permissive P-6 and P-10 are in their required state for existing plant conditions by observation of the permissive annunciator window. Quarterly surveillance shall include verification of the High Flux at Shutdown Alarm Setpoint of 1/2 decade above the existing count rate.

(10) Setpoint verification is not applicable.

(11) The TRIP ACTUATING DEVICE OPERATIONAL TEST shall include independent verification of the OPERABILITY of the undervoltage and shunt trip attachment of the Reactor Trip Breakers.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in WCAP-17070-P.

TABLE 3.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
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	2	1	2	1 2, 3, 4	17
b. Automatic Actuation Logic and Actuation Relays	2	1	2	1 2, 3, 4	14
c. Containment Pressure - High	3	2	2	1 2, 3	15
d. Pressurizer Pressure - Low	3	2	2	1 2, 3#	15
e. High Differential Pressure Between the Steam Line Header and any Steam Line	3/steam line	2/steam line in any steam line	2/steam line	1 2, 3#	15

TABLE 3.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
4. Steam Line Isolation (Continued)					
d. Steam Line Flow--High Coincident with: Steam Generator Pressure--Low	2/steam line	1/steam line in any two steam lines	1/steam line in any two steam lines	1, 2, 3	15
	1/steam generator	1/steam generator in any two steam lines	1/steam generator in any two steam lines	1, 2, 3	15
or T_{avg}--Low	1/Loop	1/loop in any two loops	1/loop in any two loops	1, 2, 3	25
5. Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2 ,3	22
b. Safety-Injection	See Item 1. above for all Safety Injection initiating functions and requirements.				
c. Steam Generator Water Level -- High-High# # # #	3/steam generator	2/steam generator in any operating steam generator	2/steam generator in any operating steam generator	1, 2	15
6. Auxiliary Feedwater# # #					
a. Automatic Actuation Logic and Actuation Relays	2	1	2	1, 2, 3	20

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	≤4.5 psig	≤4.0 psig
d. Pressurizer Pressure--Low	≥1712 psig	≥1730 psig
e. High Differential Pressure Between the Steam Line Header and any Steam Line.	≤114 psig	≤100 psi
f. Steam Line Flow--High	≤A function defined as follows: A ΔP corresponding to ~~44~~ 41.2% steam flow at 0% load increasing linearly from 20% load to a value corresponding to ~~116.5~~ 114.4% steam flow at full load	≤A function defined as follows: A ΔP corresponding to 40% (a), (b) steam flow at 0% load increasing linearly from 20% load to a value corresponding to 114% steam flow at full load

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE	TRIP SETPOINT
Coincident with: Steam Generator Pressure--Low (4)	607 ≥~~588~~ psig	(a), (b) ≥614 psig
or T_{avg}--Low	≥542.5°F	≥543°F
2. Containment Spray		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Containment Pressure--High-High Coincident with:	≤22.6 psig	≤20.0 psig
Containment Pressure--High	≤4.5 psig	≤4.0 psig
3. Containment Isolation		
a. Phase "A" Isolation		
1) Manual Initiation	N.A.	N.A.
2) Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
3) Safety Injection	See Item 1 above for all Safety Injection Allowable Values.	See Item 1 above for all Safety Injection Trip Setpoints.
b. Phase "B" Isolation		
1) Manual Initiation	N.A.	N.A.

TABLE 3.3-3 (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. psig
Containment Pressure--High	≤ 4.5 psig	≤4.0 psig
d. Steam Line Flow--High	≤A function defined as follows: A ΔP corresponding to ~~44~~ 41.2% steam flow at 0% load increasing linearly from 20% load to a value corresponding to ~~116.5~~ 114.4% steam flow at full load.	~~≤~~ 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 114% steam flow at full load.
Coincident with: Steam Line Pressure--Low (4)	≥~~588~~ 607 psig	~~≥~~ 614 psig (a), (b)
or T_{avg} -- Low	≥542.5°F	≥543°F
5. ~~Feedwater~~ Feedwater Isolation		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Safety Injection	See Item 1. for all Safety Injection Allowable Valves.	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	80.5 ≤~~81.9~~% of narrow range instrument span	(a), (b) ≤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	15.5 ≥~~8.15~~% of narrow range instrument span.	16 (a), (b) ≥~~10~~% of narrow range instrument span
c. Safety Injection	See Item 1. 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.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE#	TRIP SETPOINT
7. Loss of Power (Continued)		
b. 480V Load Centers Undervoltage		Insert "3" for all Load Centers
Load Center		
3A	[ ]	430V ±~~5~~V (10 sec ±1 sec delay)
3B	[ ]	438V ±~~5~~V (10 sec ±1 sec delay)
3C	[ ]	434V ±~~5~~V (10 sec ±1 sec delay)
3D	[ ]	434V ±~~5~~V (10 sec ±1 sec delay)
4A	[ ]	435V ±~~5~~V (10 sec ±1 sec delay)
4B	[ ]	434V ±~~5~~V (10 sec ±1 sec delay)
4C	[ ]	434V ±~~5~~V (10 sec ±1 sec delay)
4D	[ ]	430V ±~~5~~V (10 sec ±1 sec delay)
Coincident with: Safety Injection and	See Item 1. above for all Safety Injection Allowable Values.	See Item 1. above for all Safety Injection Trip Setpoints.
Diesel Generator Breaker Open	N.A.	N.A.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	ALLOWABLE VALUE#	TRIP SETPOINT
7. Loss of Power (Continued)		
c. 480V Load Centers Degraded Voltage		
Load Center		
3A	[ ]	424V ±~~5~~V (60 sec ±30 sec delay)
3B	[ ]	427V ±~~5~~V (60 sec ±30 sec delay)
3C	[ ]	437V ±~~5~~V (60 sec ±30 sec delay)
3D	[ ]	435V ±~~5~~V (60 sec ±30 sec delay)
4A	[ ]	430V ±~~5~~V (60 sec ±30 sec delay)
4B	[ ]	436V ±~~5~~V (60 sec ±30 sec delay)
4C	[ ]	434V ±~~5~~V (60 sec ±30 sec delay)
4D	[ ]	434V ±~~5~~V (60 sec ±30 sec delay)
Coincident with: Diesel Generator Breaker Open	N.A.	N.A.

Insert "3" for all Load Centers

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

TABLE 3.3-3 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

TABLE NOTATIONS

(1) Either the particulate or gaseous channel in the OPERABLE status will satisfy this LCO.

(2) Containment Gaseous Monitor Setpoint = $\frac{(3.2 \times 10^4)}{(\quad F \quad)}$ CPM,

Containment Gaseous Monitor Allowable Value = $\frac{(3.5 \times 10^4)}{(\quad F \quad)}$ CPM,

Where F = $\frac{\text{Actual Purge Flow}}{\text{Design Purge Flow (35,000 CFM)}}$

Setpoint may vary according to current plant conditions provided that the release rate does not exceed allowable limits provided in the Offsite Dose Calculation Manual.

(3) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

If no Allowable Value is specified, as indicated by [ ], the trip setpoint shall also be the allowable value.

(4) Time constants utilized in the lead-lag controller for Steam Generator Pressure- Low and Steam Line Pressure-Low are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in WCAP-17070-P.

TABLE 4.3-2

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
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.	N.A.	R	N.A.	1, 2, 3
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	M(1)	1, 2, 3(3)
c. Containment Pressure--High	N.A.	R	N.A.	N.A.	M(1)	1, 2, 3
d. Pressurizer Pressure--Low	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
e. High Differential Pressure Between the Steam Line Header and any Steam Line	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
f. Steam Line Flow--High Coincident with:	S	R	Q(5)	N.A	N.A	1, 2, 3(3)
Steam Generator Pressure--Low	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
or						
Tavg--Low	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)

(a), (b)

(a),(b)

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
4. Steamline Isolation (Continued)						
c. Containment Pressure--High-High	N.A.	R	N.A.	R	M(1)	1, 2, 3
Coincident with: Containment Pressure--High	N.A.	R (a),(b)	N.A (a),(b)	R	M(1)	1, 2, 3
d. Steam Line Flow--High	S(3)	R	Q(5)	N.A.	N.A.	1, 2, 3
Coincident with: Steam Generator Pressure--Low	S(3)	R	Q(5)	N.A.	N.A.	1, 2, 3
or Tavg--Low	S(3)	R	Q(5)	N.A.	N.A.	1, 2, 3
5. Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	1, 2
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Steam Generator Water Level--High-High	S	R (a),(b)	Q (a),(b)	N.A.	N.A.	1, 2 , 3
6. Auxiliary Feedwater (2)						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	R	1, 2, 3
b. Steam Generator Water Level--Low-Low	S	R (a),(b)	Q (a),(b)	N.A.	N.A.	1, 2, 3

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

CHANNEL FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL CALIBRATION	ANALOG CHANNEL OPERATIONAL TEST	TRIP ACTUATING DEVICE OPERATIONAL TEST	ACTUATION LOGIC TEST #	MODES FOR WHICH SURVEILLANCE IS REQUIRED
8. Engineering Safety Features Actuation System Interlocks						
a. Pressurizer Pressure	N.A.	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
b. Tavg--Low	N.A.	R	Q(5)	N.A.	N.A.	1, 2, 3(3)
9. Control Room Ventilation Isolation						
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.	N.A.	N.A.	N.A.	
b. Safety Injection	See Item 1. above for all Safety Injection Surveillance Requirements.					
c. Containment Radioactivity--High	S	R	M	N.A.	N.A.	(4)
d. Containment Isolation Manual Phase A or Manual Phase B	N.A.	N.A.	N.A.	R	N.A.	1, 2, 3, 4
e. Control Room Air Intake Radiation Level	S	R	M	N.A.	N.A.	All

move to new page 3-34a

TABLE NOTATIONS

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.
(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.
(3) The provisions of Specification 4.0.4 are not applicable for entering Mode 3, provided that the applicable surveillances are completed within 96 hours from entering Mode 3.
(4) Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment.
(5) Test of alarm function not required when alarm locked in.

#At least once per 18 months each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay.

New Page 3/4 3-34a

TABLE 4.3-2 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

add notes (a), (b)

TABLE NOTATIONS

At least once per 18 months each Actuation Logic Test shall include energization of each relay and verification of OPERABILITY of each relay.

(a) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(b) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Nominal Trip Setpoint (NTS) at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the NTS are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field settings) to confirm channel performance. The NTS and methodologies used to determine the as-found and the as-left tolerances are specified in WCAP-17070-P.

(1) Each train shall be tested at least every 62 days on a STAGGERED TEST BASIS.

(2) Auxiliary feedwater manual initiation is included in Specification 3.7.1.2.

(3) The provisions of Specification 4.0.4 are not applicable for entering Mode 3, provided that the applicable surveillances are completed within 96 hours from entering Mode 3.

(4) Applicable in MODES 1, 2, 3, 4 or during CORE ALTERATIONS or movement of irradiated fuel within the containment.

(5) Test of alarm function not required when alarm locked in.

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

2465

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE* with a lift setting of ~~2485~~ psig + 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.

* 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.

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

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of ~~2485~~ 2465 psig + 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.

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

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

Replace with the next page

MATERIAL PROPERTY BASIS

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

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

FPL 32 EFPY HEATUP CURVES

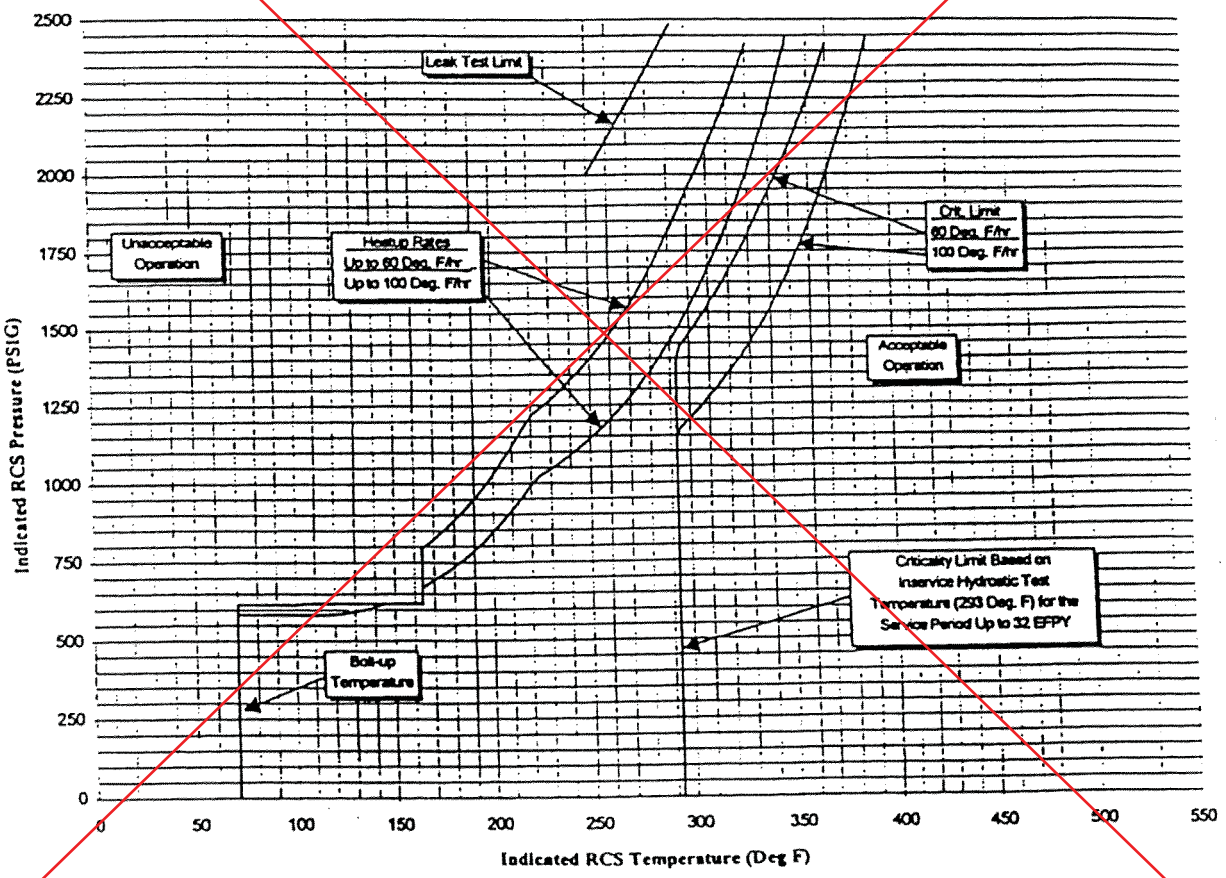


FIGURE 3.4-2 Turkey Point Units 3 and 4 Reactor Coolant System Heatup Limitations (Heatup Rate of 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

INSERT THIS FIGURE
FOR FIGURE 3.4-2

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: Intermediate/Lower Shell Circumferential Weld Seam Ht. # 71249 and Upper Shell Forging

LIMITING ART VALUES AT 48 EFPY: 1/4T, 231°F (Circ Flaw ART), 141°F (Axial Flaw ART)
3/4T, 192°F (Circ Flaw ART), 124°F (Axial Flaw ART)

FPL / FLA 48 EFPY HEATUP CURVES

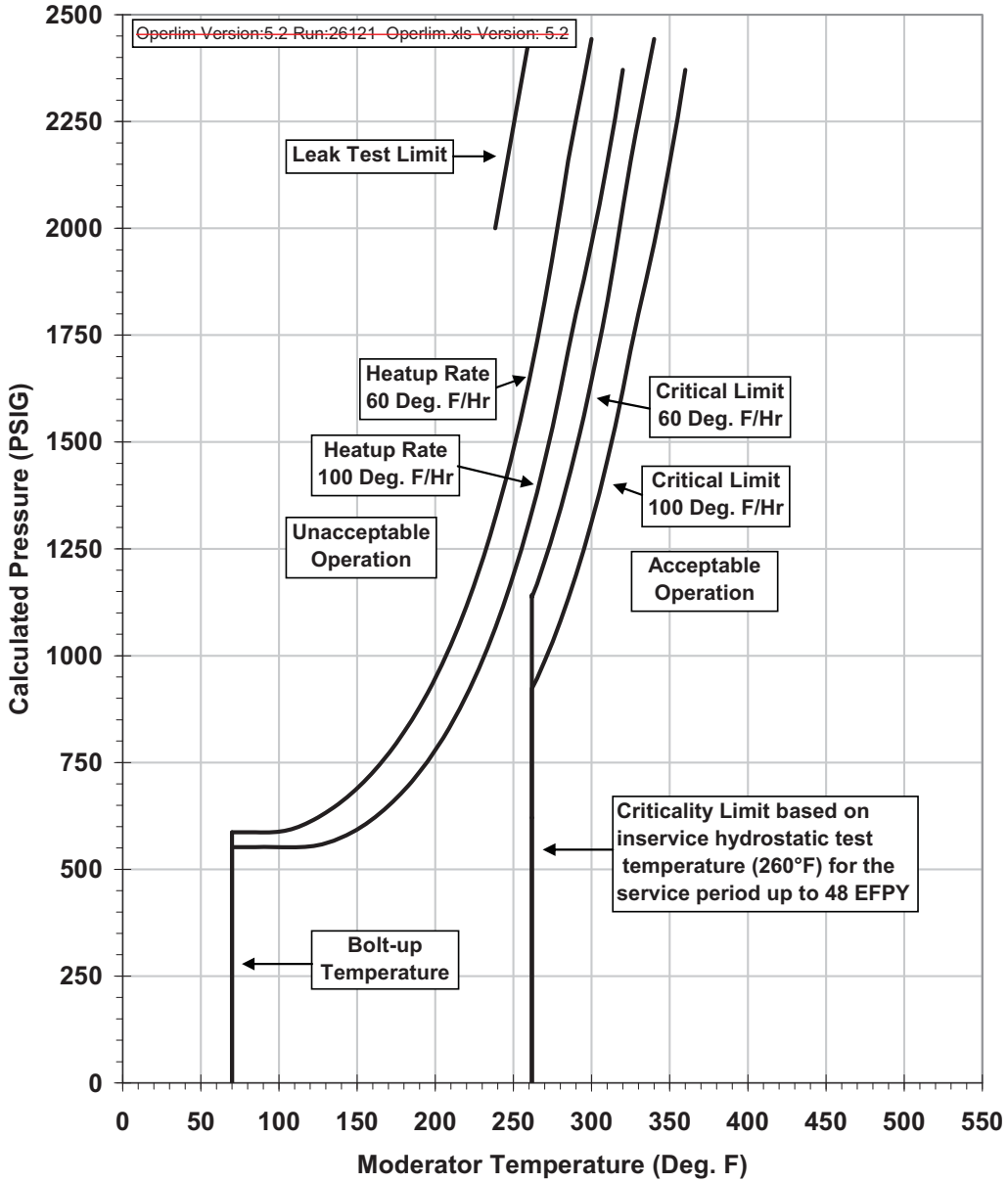


FIGURE 3.4-2 Turkey Point Units 3 and 4 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 48 EFPY (Without Margins for Instrumentation Errors)

Replace with the next page

MATERIAL PROPERTY BASIS

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

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

FPL 32 EFPY COOLDOWN CURVES

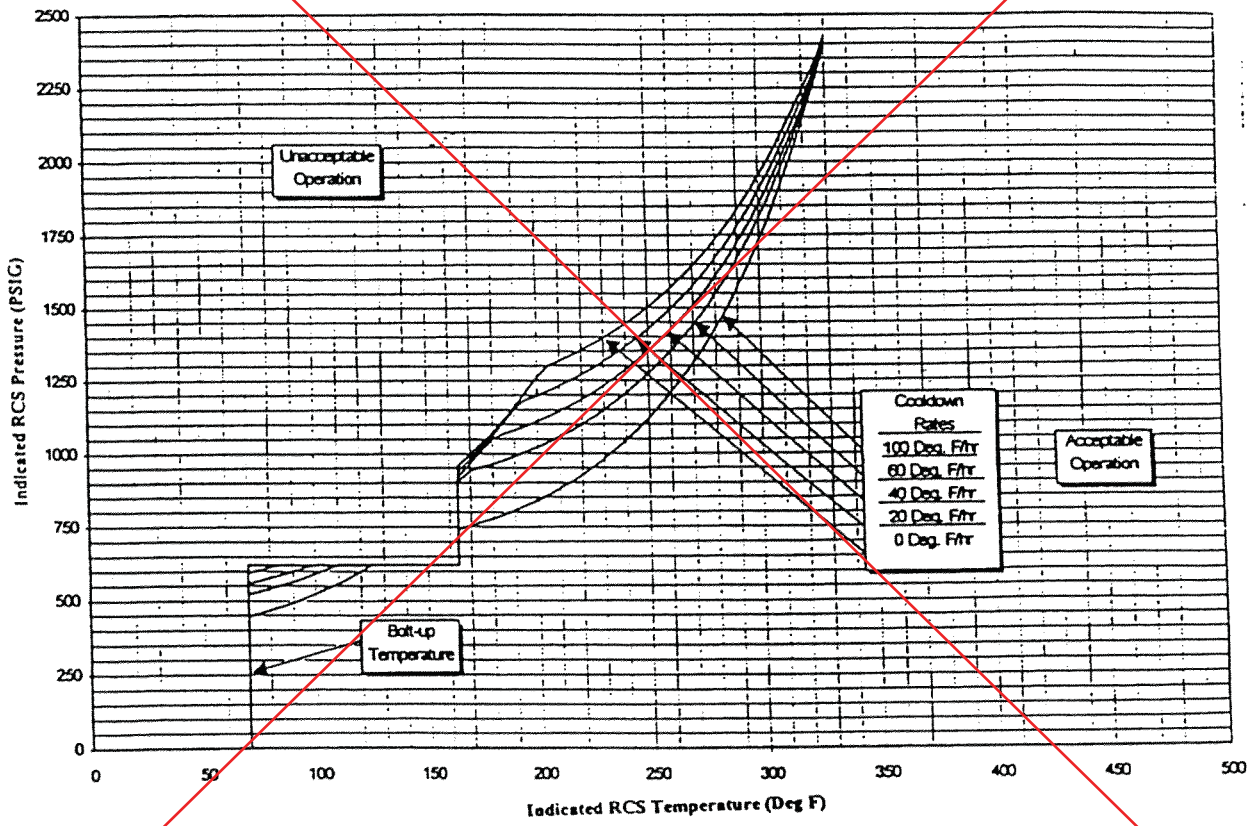


FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rate of 0, 20, 40, 60 and 100°F/hr) Applicable for 32 EFPY (Without Margins for Instrumentation Errors)

INSERT THIS FIGURE
FOR FIGURE 3.4-3

MATERIAL PROPERTY BASIS

LIMITING MATERIALS: Intermediate/Lower Shell Circumferential Weld Seam Ht. # 71249 and Upper Shell Forging

LIMITING ART VALUES AT 48 EFPY: 1/4T, 231°F (Circ Flaw ART), 141°F (Axial Flaw ART)
3/4T, 192°F (Circ Flaw ART), 124°F (Axial Flaw ART)

FPL / FLA 48 EFPY COOLDOWN CURVES

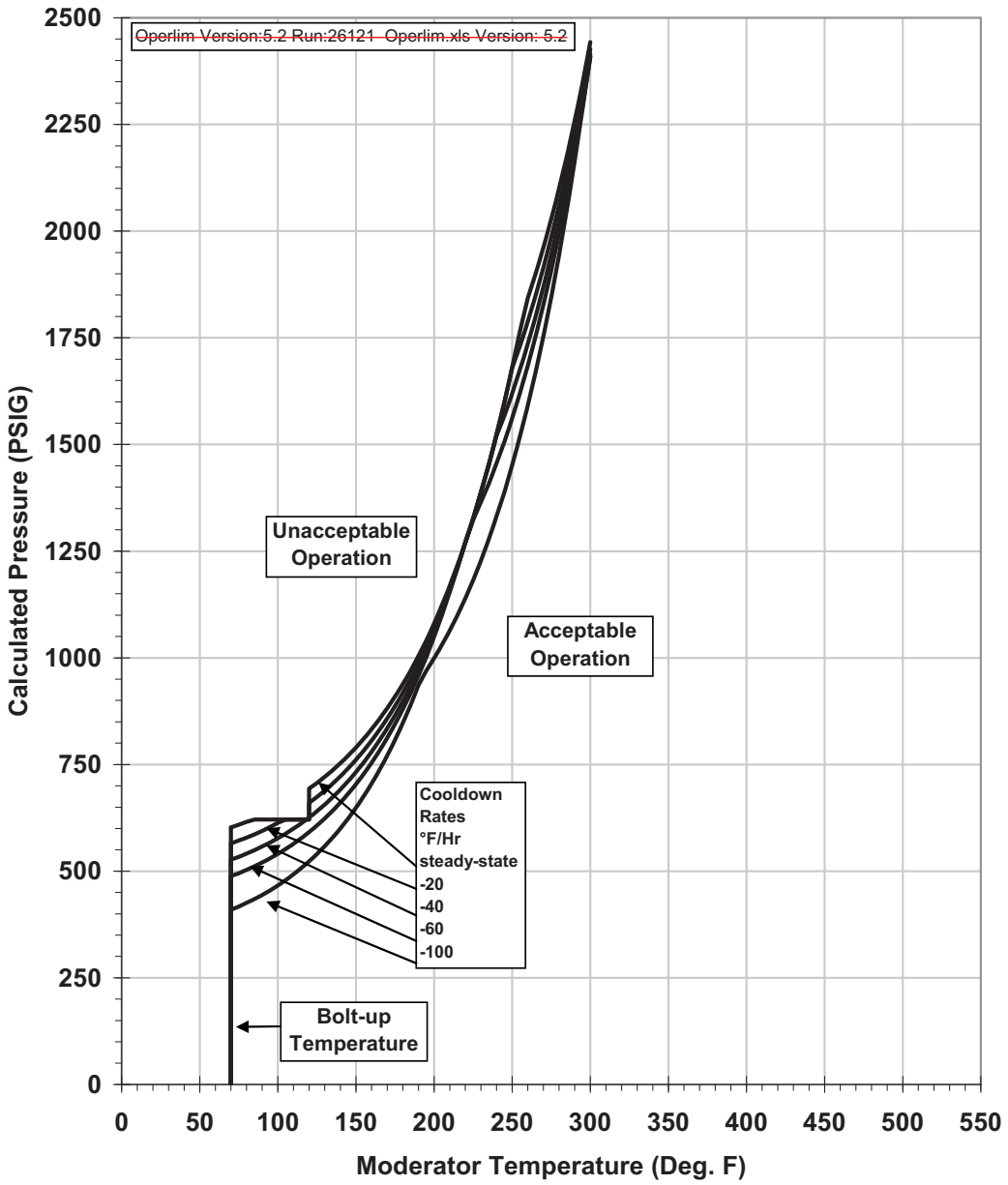


FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for 48 EFPY (Without Margins for Instrumentation Errors)

REACTOR COOLANT SYSTEM

OVERPRESSURE MITIGATING SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.9.3 The high pressure safety injection flow paths to the Reactor Coolant System (RCS) shall be isolated, and at least one of the following Overpressure Mitigating Systems shall be OPERABLE:

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

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

APPLICABILITY MODES 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), 5, and 6 with the reactor vessel head on.

ACTION:

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

b. With one PORV inoperable in MODE 4 (when the temperature of any RCS cold leg is less than or equal to 275°F), restore the inoperable PORV to OPERABLE status within 7 days or depressurize and vent the RCS through at least a 2.20 square inch vent within the next 8 hours.

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

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

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

f. The provisions of Specification 3.0.4 are not applicable.

No change on this page. Included to show context of change on next page

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each Reactor Coolant System (RCS) accumulator shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3*.

ACTION:

a. With one accumulator inoperable, except as a result of boron concentration not being within limits, restore the inoperable accumulator to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

b. With one accumulator inoperable due to the boron concentration not being within the limits, restore boron concentration back to the required limits within 72 hours, or be in at least HOT STANDBY within 6 hours and reduce pressurizer pressure to less than 1000 psig within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.5.1.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1) Verifying the borated water volume in each accumulator is between 6520 and 6820 gallons, and

2) Verifying that the nitrogen cover pressure in each accumulator is between 600 and 675 psig, and

3) Verifying that each accumulator isolation valve is open by control room indication (power may be restored to the valve operator to perform this surveillance if redundant indicator is inoperable).

*Pressurizer pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the solution in the water-filled accumulator is between ~~1950~~ 2300 and ~~2350~~ 2600 ppm;

c. At least once per 31 days, when the RCS pressure is above 1000 psig, by verifying that the power to the isolation valve operator is disconnected by a locked open breaker.

d. At least once per 18 months, each accumulator check valve shall be checked for operability.

3/4.5.4 REFUELING WATER STORAGE TANK

LIMITING CONDITION FOR OPERATION

3.5.4 For single Unit operation, one refueling water storage tank (RWST) shall be OPERABLE or for dual Unit operation two RWSTs shall be OPERABLE with:

a. A minimum indicated borated water volume of 320,000 gallons per RWST,

b. A ~~minimum~~ boron concentration ~~of 1950 ppm of boron~~, between 2400 ppm and 2600 ppm

c. A minimum solution temperature of 39°F, and

d. A maximum solution temperature of 100°F.

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

ACTION:

With less than the required number of RWST(s) OPERABLE, restore the tank(s) to OPERABLE status within 1 hour or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.5.4 The required RWST(s) shall be demonstrated OPERABLE:

a. At least once per 7 days by:

1) Verifying the indicated borated water volume in the tank, and

2) Verifying the boron concentration of the water.

b. By verifying the RWST temperature is within limits whenever the outside air temperature is less than 39°F or greater than 100°F at the following frequencies:

1) Within one hour upon the outside temperature exceeding its limit for consecutive 23 hours, and

2) At least once per 24 hours while the outside temperature exceeds its limit.

No change on this page. Included to show context of change on next page

3/4.7 PLANT SYSTEMS

3/4.7.1 TURBINE CYCLE

SAFETY VALVES

LIMITING CONDITION FOR OPERATION

3.7.1.1 All main steam line Code safety valves associated with each steam generator shall be OPERABLE with lift settings as specified in Table 3.7-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTION:

With (3) reactor coolant loops and associated steam generators in operation and with one or more main steam line Code safety valves inoperable, and

a. in MODES 1 and 2, with a positive Moderator Temperature Coefficient, operation may continue provided that, within 4 hours, either the inoperable valve(s) are restored to OPERABLE status or the Power Range Neutron Flux High Trip Setpoint is reduced to the maximum allowable percent of RATED THERMAL POWER listed In Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours, or

b. in MODES 1 and 2, with a negative or zero Moderator Temperature Coefficient; or in Mode 3, with a positive, negative or zero Moderator Temperature Coefficient, operation may continue provided that, within 4 hours, either the inoperable valve(s) are restored to OPERABLE status or reactor power is reduced to less than or equal to the maximum allowable percent of RATED THERMAL POWER listed in Table 3.7-1; otherwise, be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 12 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.1 No additional requirements other than those required by Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODE 3.

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	~~53~~ ← 44
2	~~33~~ ← 27
3	~~14~~ ← 10

TABLE 3.7-2

STEAM LINE SAFETY VALVES PER LOOP

	VALVE NUMBER			LIFT SETTING (±3%)* **	ORIFICE SIZE SQUARE INCHES
	Loop A	Loop B	Loop C		
1.	RV1400	RV1405	RV1410	1085 psig	16
2.	RV1401	RV1406	RV1411	1100 psig	16
3.	RV1402	RV1407	RV1412	1105 → ~~1115~~ psig	16
4.	RV1403	RV1408	RV1413	1105 → ~~1130~~ psig	16

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

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

STANDBY FEEDWATER SYSTEM

145,000

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two Standby Steam Generator Feedwater Pumps shall be OPERABLE* and at least ~~135,000~~ gallons of water (indicated volume), shall be in the Demineralized Water Storage Tank.**

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.

145,000

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

145,000

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 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.

This is a new Technical Specification

PLANT SYSTEMS

3/4.7.1.7 FEEDWATER ISOLATION

LIMITING CONDITION FOR OPERATION

3.7.1.7 Six Feedwater Control Valves (FCVs) both main and bypass and six Feedwater Isolation Valves (FIVs) both main and bypass shall be OPERABLE*.

APPLICABILITY: MODES 1, 2 and 3**

ACTION:

a. With one or more FCVs inoperable, restore operability, or close or isolate the inoperable FCVs within 72 hours and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one or more FIVs inoperable, restore operability, or close or isolate the inoperable FIV(s) within 72 hours and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

c. With one or more bypass valves in different steam generator flow paths inoperable, restore operability, or close or isolate the inoperable bypass valve(s) within 72 hours and verify that the inoperable valve(s) is closed or isolated at least once per 7 days or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

d. With two valves in the same steam generator flow path inoperable, restore operability, or isolate the affected flowpath within 8 hours or be in HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

SURVEILLANCE REQUIREMENTS

4.7.1.7 Each FCV, FIV and bypass valve shall be demonstrated OPERABLE:

a. At least every 18 months by:

   1) Verifying that each FCV, FIV and bypass valve actuates to the isolation position on an actual or simulated actuation signal.

b. In accordance with the Inservice Testing Program by:

   1) Verifying that each FCV, FIV and bypass valve isolation time is within limits.

---

* Separate Condition entry is allowed for each valve.
** The provisions of specification 3.0.4 and 4.0.4 are not applicable

SURVEILLANCE REQUIREMENTS (Continued)

4.8.1.1.2 Each diesel generator shall be demonstrated OPERABLE*:

a. At least once per 31 days on a STAGGERED TEST BASIS by:

1) Verifying the fuel volume in the day and skid-mounted fuel tanks (Unit 4-day tank only),

2) Verifying the fuel volume in the fuel storage tank,

3) Verifying the lubricating oil inventory in storage,

4) Verifying the diesel starts and accelerates to reach a generator voltage and frequency of ~~4160 ± 420~~ 3950-4350 volts and 60 ± ~~1.2~~ 0.6 Hz. Once per 184 days, these conditions shall be reached within 15 seconds after the start signal from normal conditions. For all other starts, warmup procedures, such as idling and gradual acceleration as recommended by the manufacturer may be used. The diesel generator shall be started for this test by using one of the following signals:

a) Manual, or

b) Simulated loss-of-offsite power by itself, or

c) Simulated loss-of-offsite power in conjunction with an ESF Actuation test signal, or

d) An ESF Actuation test signal by itself.

5) Verifying the generator is synchronized, loaded** to 2300 - 2500 kW (Unit 3), 2650-2850 kW (Unit 4)***, operates at this loaded condition for at least 60 minutes and for Unit 3 until automatic transfer of fuel from the day tank to the skid mounted tank is demonstrated, and the cooling system is demonstrated OPERABLE.

6) Verifying the diesel generator is aligned to provide standby power to the associated emergency busses.

* All diesel generator starts for the purpose of these surveillances may be proceeded by a prelube period as recommended by the manufacturer.

** May include gradual loading as recommended by the manufacturer so that the mechanical stress and wear on the diesel engine is minimized.

*** Momentary transients outside these load bands do not invalidate this test.

SURVEILLANCE REQUIREMENTS (Continued)

b. Demonstrating at least once per 92 days that a fuel transfer pump starts automatically and transfers fuel from the storage system to the day tank,

c. At least once per 31 days and after each operation of the diesel where the period of operation was greater than or equal to 1 hour by checking for and removing accumulated water from the day and skid-mounted fuel tanks (Unit 4-day tank only);

d. At least once per 31 days by checking for and removing accumulated water from the fuel oil storage tanks;

e. By verifying fuel oil properties of new fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.

f. By verifying fuel oil properties of stored fuel oil are tested in accordance with, and maintained within the limits of, the Diesel Fuel Oil Testing Program.

g. At least once per 18 months, during shutdown (applicable to only the two diesel generators associated with the unit):

1) Deleted

2)* Verifying the generator capability to reject a load of greater than or equal to 380 kw while maintaining voltage at ~~4160 ± 420~~ 3950-4350 volts and frequency at 60 ~~±1.2~~ 0.6 Hz;

3)* Verifying the generator capability to reject a load of greater than or equal to 2500 kW (Unit 3), 2874 kW (Unit 4) without tripping. The generator voltage shall return to less than or equal to 4784 volts within 2 seconds following the load rejection;

4) Simulating a loss-of-offsite power by itself, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses, and

b. Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently

* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

3950-4350 0.6

connected loads within 15 seconds, energizes the auto–connected shutdown loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the auto-connected shutdown loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~4160 ± 420~~ volts and 60 ± ~~1.2~~ Hz during this test.

0.6 3950-4350

5) Verifying that on an ESF Actuation test signal, without loss-of-offsite power, the diesel generator starts on the auto-start signal and operates on standby for greater than or equal to 5 minutes. The generator voltage and frequency shall be ~~4160 ± 420~~ volts and 60 ± ~~1.2~~ Hz within 15 seconds after the auto-start signal; the steady-state generator voltage and frequency shall be maintained within these limits during this test;

6) Simulating a loss-of-offsite power in conjunction with an ESF Actuation test signal, and:

a) Verifying deenergization of the emergency busses and load shedding from the emergency busses;

b) Verifying the diesel starts on the auto-start signal, energizes the emergency busses with any permanently connected loads within 15 seconds, energizes the auto-connected emergency (accident) loads through the load sequencer and operates for greater than or equal to 5 minutes while its generator is loaded with the emergency loads. After automatic load sequencing, the steady-state voltage and frequency of the emergency busses shall be maintained at ~~4160 ± 420~~ volts and 60 ± ~~1.2~~ Hz during this test; and

3950-4350 0.6

c) Verifying that diesel generator trips that are made operable during the test mode of diesel operation are inoperable.

7)* # Verifying the diesel generator operates for at least 24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to 2550-2750 kW (Unit 3), 2950-3150 kW (Unit 4)** and during the remaining 22 hours of this test, the diesel generator shall be loaded to 2300-2500 kW (Unit 3), 2650-2850 kW (Unit 4)**. The generator voltage and frequency shall be ~~4160 ± 420~~ volts and 60 ± ~~1.2~~ Hz within 15 seconds after the start signal; the steady-state generator voltage and frequency

3950-4350 0.6

* For the purpose of this test, warmup procedures, such as idling, gradual acceleration, and gradual loading as recommended by the manufacturer may be used.

** Momentary transients outside these load bands do not invalidate this test.

This test may be performed during POWER OPERATION

shall be maintained within these limits during this test. Within 5 minutes after completing this 24-hour test, verify the diesel starts and accelerates to reach a generator voltage and frequency of ~~4160 ± 420~~ 3950-4350 volts and 60 ± ~~1.2~~ 0.6 Hz within 15 seconds after the start signal.**

8) Verifying that the auto-connected loads to each diesel generator do not exceed 2500 kW (Unit 3), 2874 kW (Unit 4);

9) Verifying the diesel generator's capability to:

   a) Synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   b) Transfer its loads to the offsite power source, and

   c) Be restored to its standby status.

10) Verifying that the diesel generator operating in a test mode, connected to its bus, a simulated Safety Injection signal overrides the test mode by: (1) returning the diesel generator to standby operation, and (2) automatically energizing the emergency loads with offsite power;

11) Verifying that the fuel transfer pump transfers fuel from the fuel storage tank (Unit 3), fuel storage tanks (Unit 4) to the day tanks of each diesel associated with the unit via the installed cross-connection lines;

12) Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ± 10% of its design interval;

13) Verifying that the diesel generator lockout relay prevents the diesel generator from starting;

** If verification of the diesel's ability to restart and accelerate to a generator voltage and frequency of ~~4160 ± 420~~ 3950-4350 volts and 60 ± ~~1.2~~ 0.6 Hz within 15 seconds following the 24 hour operation test of Specification 4.8.1.1.2.g.7) is not satisfactorily completed, it is not necessary to repeat the 24-hour test. Instead, the diesel generator may be operated between 2300-2500 kW Unit 3, 2650-2850 kW (Unit 4) for 2 hours or until operating temperature has stabilized (whichever is greater). Following the 2 hours/operating temperature stabilization run, the EDG is to be secured and restarted within 5 minutes to confirm its ability to achieve the required voltage and frequency within 15 seconds.

h. At least once per 10 years or after any modifications which could affect diesel generator interdependence by starting all required diesel generators simultaneously and verifying that all required diesel generators provide 60 ~~±1.2~~ 0.6 Hz frequency and ~~4160 ± 420~~ 3950-4350 volts in less than or equal to 15 seconds: and

I. At least once per 10 years by:

1) Draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.*

2) For Unit 4 only, performing a pressure test of those portions of the diesel fuel oil system designed to Section III, subsection ND of the ASME Code in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda.

4.8.1.1.3 Reports - (Not Used)

* A temporary Class III fuel storage system containing a minimum volume of 38,000 gallons of fuel oil may be used for up to 10 days during the performance of Surveillance Requirement 4.8.1.1.2i.1 for the Unit 3 storage tank while Unit 3 is in Modes 5. 6. or defueled. If the diesel fuel oil storage tank is not returned to service within 10 days. Technical Specification 3.8.1.1 Action b and 3.8.1.2 Action apply to Unit 4 and Unit 3 respectively.

3/4.9 REFUELING OPERATIONS

3/4.9.1 BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 The boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met; either:

a. A K_{eff} of 0.95 or less, or

b. A boron concentration of greater than or equal to ~~1950 ppm~~ 2300.

APPLICABILITY: MODE 6.*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at greater than or equal to 16 gpm of a solution containing greater than or equal to 3.0 wt% (5245 ppm) boron or its equivalent until K_{eff} is reduced to less than or equal to 0.95 or the boron concentration is restored to greater than or equal to ~~1950~~ 2300 ppm, whichever is the more restrictive.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full-length control rod in excess of 3 feet from its fully inserted position within the reactor vessel.

4.9.1.2 The boron concentration of the Reactor Coolant System and the refueling canal shall be determined by chemical analysis at least once per 72 hours.

4.9.1.3 Valves isolating unborated water sources** shall be verified closed and secured in position by mechanical stops or by removal of air or electrical power at least once per 31 days.

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* The reactor shall be maintained in MODE 6 whenever fuel is in the reactor vessel with the vessel head closure bolts less than fully tensioned or with the head removed.

** The primary water supply to the boric acid blender may be opened under administrative controls for makeup.

3/4.9.14 SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.14 The following conditions shall apply to spent fuel storage:

a. The minimum boron concentration in the Spent Fuel Pit shall be ~~1950~~ 2300 ppm.

b. The combination of initial enrichment, burnup, and cooling time of each fuel assembly stored in the Spent Fuel Pit shall be in accordance with Specification 5.5.1.

APPLICABILITY: At all times when fuel is stored in the Spent Fuel Pit.

ACTION:

a. With boron concentration in the Spent Fuel Pit less than ~~1950~~ 2300 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to ~~1950~~ 2300 ppm or greater.

b. With condition b not satisfied, suspend movement of additional fuel assemblies into the Spent Fuel Pit and restore the spent fuel storage configuration to within the specified conditions.

c. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.14.1 The boron concentration of the Spent Fuel Pit shall be verified to be ~~1950~~ 2300 ppm or greater at least once per month.

4.9.14.2 A representative sample of inservice Metamic inserts shall be visually inspected in accordance with the Metamic Surveillance Program described in UFSAR Section 16.2. The surveillance program ensures that the performance requirements of Metamic are met over the surveillance interval.

5.5 FUEL STORAGE

5.5.1 CRITICALITY

5.5.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A k_{eff} less than 1.0 when flooded with unborated water, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

b. A k_{eff} less than or equal to 0.95 when flooded with water borated to ~~650~~ 500 ppm, which includes an allowance for biases and uncertainties as described in UFSAR Chapter 9.

c. A nominal 10.6 inch center-to-center distance for Region I and 9.0 inch center-to-center distance for Region II for the two region spent fuel pool storage racks. A nominal 10.1 inch center-to-center distance in the east-west direction and a nominal 10.7 inch center-to-center distance in the north-south direction for the ~~Region I~~ cask area storage rack.

d. A maximum enrichment loading for fuel assemblies of 4.5 weight percent of U-235.

e. No restriction on storage of fresh or irradiated fuel assemblies in the cask area storage rack.

f. Fresh or irradiated fuel assemblies not stored in the cask area storage rack shall be stored in accordance with Specification 5.5.1.3 or configurations that have been shown to comply with Specification 5.5.1.1a and 5.5.1.1b using the NRC approved methodology in UFSAR Chapter 9.

5.5.1.2 The racks for new fuel storage are designed to store fuel in a safe subcritical array and shall be maintained with:

a. A nominal 21 inch center-to-center spacing to assure k_{eff} equal to or less than 0.98 for optimum moderation conditions and equal to or less than 0.95 for fully flooded conditions.

b. ~~Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.5 weight percent of U-235.~~

Fuel assemblies placed in the New Fuel Storage Area shall contain no more than 4.7 weight percent of U-235 if the assembly contains no burnable absorber rods and no more than 5.0 weight percent of U-235 if the assembly contains at least 16 IFBA rods (or an equivalent amount of other burnable absorber).

Any 2x2 array of Region II storage cells may adjoin assemblies that are located in the outer row adjacent to the spent fuel pit wall. The outer row of fuel assemblies facing the pool wall need not contain any Metamic inserts or full length RCCAs, as long as the fuel is loaded to comply with the allowable storage patterns defined in Figure 5.5-4.

DESIGN FEATUR

5.5.1.3 Credit for burnup and cooling time is taken in determining acceptable placement locations for spent fuel in the two-region spent fuel racks. Fresh or irradiated fuel assemblies shall be stored in compliance with the following:

a. Any 2x2 array of Region I storage cells containing fuel shall comply with the storage patterns in Figure 5.5-1 and the requirements of Table 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less than that shown for the 2x2 array.

reactive

b. Any 2x2 array of Region II storage cells containing fuel shall:

i. Comply with the storage patterns in Figure 5.5-2 and the requirements of Table 5.5-1 and 5.5-2, as applicable. The reactivity rank of fuel assemblies in the 2x2 array (rank determined using Table 5.5-3) shall be equal to or less than that shown for the 2x2 array,

reactive

ii. Have the same directional orientation for Metamic inserts in a contiguous group of 2x2 arrays where Metamic inserts are required,

iii. Comply with the requirements of 5.5.1.3.c for cells adjacent to Region I racks, and

iv. Comply with the requirements of 5.5.1.3.d for cells adjacent to the spent fuel pit walls.

storage cells

c. Any 2x2 array of Region II storage cells that interface with Region I shall comply with the rules of Figure 5.5-3. Arrays II-E and II-F may interface with Region I without special restriction.

d. ~~Any 2x2 array of Region II storage cells may adjoin a row of assemblies with a reactivity rank of II-2 (or lower) that is located in the outer row adjacent to the spent fuel pit wall. The outer row of reactivity rank II-2 (or lower) fuel assemblies need not contain any Metamic inserts or full length RCCAs, as long as the following additional requirements are met:~~

~~i. Fuel is loaded to comply with the allowable storage patterns defined in Figure 5.5-4, and~~

~~ii. Arrays II-E and II-F are loaded without any additional restriction on that 2x2 array. Arrays II-E and II-F do not have empty cells, Metamic inserts, or RCCAs that restrict the interface with the adjoining reactivity rank II-2 (or lower) fuel assemblies.~~

DRAINAGE

5.5.2 The spent fuel storage pit is designed and shall be maintained to prevent inadvertent draining of the pool below a level of 6 feet above the fuel assemblies in the storage racks.

CAPACITY

5.5.3 The spent fuel pool storage racks are designed and shall be maintained with a storage capacity limited to no more than 1404 fuel assemblies in two region storage racks, and the cask area storage rack is designed and shall be maintained with a storage capacity limited to no more than 131 fuel assemblies. The total spent fuel pool storage capacity is limited to no more than 1535 fuel assemblies.

REPLACE TABLE 5.5-1 WITH THE NEXT TWO PAGES

Table 5.5-1

Blanketed Fuel - Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See note 1 for use of Table 5.5-1

Fuel Category	Blanketed Fuel Storage Curve Coefficients[1]							Blanketed Fuel Minimum Burnup[1] (GWd/MTU) for Initial Enrichment[2]					
	A	B	C	D	E	F	G	Cooling Time[3]	2.5 w%	3.0 w%	3.3 w%	4.0 w%	4.5 w%
I-1[4]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
I-2	18.8602	-1.090486	0.266387	-0.00474496	-0.158563	0.00314739	-30.1637	0	10.17	16.60	20.20	27.83	32.62
								2.5	9.87	16.11	19.59	26.96	31.57
								5	9.60	15.67	19.06	26.19	30.62
								10	9.18	14.98	18.20	24.94	29.10
								15	8.92	14.52	17.62	24.08	28.04
								20	8.82	14.30	17.32	23.61	27.45
II-1	16.2639	-0.712257	0.175883	-0.00399237	-0.166686	0.00370969	-19.5118	0	16.70	22.87	26.40	34.15	39.25
								2.5	16.13	22.10	25.52	32.99	37.90
								5	15.62	21.43	24.74	31.96	36.70
								10	14.82	20.34	23.49	30.32	34.78
								15	14.27	19.61	22.65	29.23	33.50
								20	13.99	19.24	22.22	28.67	32.85
II-2	14.4600	-0.372732	0.132275	-0.00617104	-0.187813	0.00526411	-12.8293	0	20.99	27.20	30.83	39.05	44.69
								2.5	20.19	26.18	29.68	37.59	43.02
								5	19.48	25.28	28.67	36.32	41.57
								10	18.32	23.85	27.07	34.35	39.32
								15	17.50	22.89	26.04	33.11	37.94
								20	17.04	22.42	25.56	32.62	37.44
II-3	15.4624	-0.501267	-0.06553	0.00160009	-0.161078	0.00340497	-11.2483	0	24.27	30.63	34.32	42.58	48.18
								2.5	23.17	29.33	32.91	40.90	46.31
								5	22.19	28.18	31.65	39.41	44.65
								10	20.60	26.32	29.63	37.00	41.97
								15	19.53	25.05	28.25	35.36	40.13
								20	18.96	24.38	27.51	34.47	39.14

Table 5.5-1 (continued)

Blanketed Fuel - Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See note 1 for use of Table 5.5-1

Fuel Category	Blanketed Fuel Storage Curve Coefficients[1]							Blanketed Fuel Minimum Burnup[1] (GWd/MTU) for Initial Enrichment[2]					
	A	B	C	D	E	F	G	Cooling Time[3]	2.5 w%	3.0 w%	3.3 w%	4.0 w%	4.5 w%
II-4	15.3172	-0.444842	-0.114363	0.00273060	-0.162664	0.00344467	-9.1868	0	26.33	32.76	36.52	44.96	50.73
								2.5	25.09	31.34	34.98	43.16	48.73
								5	24.00	30.08	33.61	41.55	46.96
								10	22.25	28.04	31.41	38.97	44.09
								15	21.06	26.67	29.92	37.20	42.14
								20	20.44	25.94	29.13	36.27	41.10
II-5	15.1701	-0.387768	-0.163521	0.00394514	-0.164014	0.00345174	-7.1273	0	28.37	34.89	38.71	47.35	53.29
								2.5	27.02	33.34	37.05	45.41	51.15
								5	25.82	31.97	35.57	43.69	49.26
								10	23.90	29.77	33.20	40.93	46.22
								15	22.60	28.28	31.59	39.05	44.14
								20	21.93	27.50	30.75	38.06	43.05
II-6	13.4516	-0.078364	-0.266734	0.00288411	-0.147006	0.00446530	-3.3460	0	29.79	36.30	40.19	49.21	55.60
								2.5	28.30	34.64	38.42	47.20	53.42
								5	26.97	33.17	36.87	45.45	51.53
								10	24.86	30.85	34.43	42.73	48.61
								15	23.44	29.35	32.88	41.05	46.85
								20	22.73	28.66	32.20	40.41	46.23
II-7	13.7900	-0.086680	-0.355570	0.00574698	-0.145745	0.00426994	-2.0705	0	31.86	38.52	42.49	51.70	58.23
								2.5	30.17	36.65	40.53	49.50	55.86
								5	28.67	35.02	38.81	47.58	53.80
								10	26.31	32.45	36.11	44.60	50.61
								15	24.76	30.80	34.41	42.76	48.67
								20	24.03	30.09	33.70	42.06	47.99

~~Table 5.5-1 (continued)~~

Blanketed Fuel - Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See note 1 for use of Table 5.5-1

Fuel Category	Blanketed Fuel Storage Curve Coefficients[1]							Blanketed Fuel Minimum Burnup[1] (GWd/MTU) for Initial Enrichment[2]					
	A	B	C	D	E	F	G	Cooling Time[3]	2.5 w%	3.0 w%	3.3 w%	4.0 w%	4.5 w%
II-8	14.1212	-0.094016	-0.448138	0.00877894	-0.143511	0.00402944	-0.7808	0	33.93	40.74	44.80	54.20	60.86
								2.5	32.04	38.67	42.63	51.80	58.29
								5	30.37	36.86	40.74	49.71	56.06
								10	27.75	34.04	37.79	46.47	52.61
								15	26.07	32.25	35.94	44.47	50.51
								20	25.34	31.51	35.19	43.71	49.75

Notes

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" given in the table for the assembly "cooling time" and "initial enrichment." Alternatively, the specific minimum burnup required for each fuel assembly may be calculated from the following equation: $Bu = A \times En + B \times En^2 + C \times Ct + D \times Ct^2 + E \times Ct \times En + F \times Ct^2 \times En + G$. Only cooling times of 0, 2.5, 5, 10, 15 and 20 years may be used in this equation. Actual cooling time (Ct) is rounded down to the nearest value.

2. Nominal central zone U-235 enrichment: Axial blanket material is not considered when determining enrichment.

3. Cooling time in years.

4. Fresh unburned fuel up to 4.5 w% U-235 enrichment: No burnup is required.

INSERTS FOR TABLE 5.5-1
(PAGE 1 of 2)

Table 5.5-1

Blanketed Fuel – Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En)[2] and Cooling Time (Ct)[3]

See notes 1-5 for use of Table 5.5-1

	Fuel Category[4,5]					
Coefficients	**I-3**	**II-1**	**II-2a**	**II-2b**	**II-2c**	**II-3**
A1	3.94	-36.24	-25.34	-16.44	-4.202	21.29
A2	-6.213	33.85	24.57	16.93	6.41	-17.14
A3	2.867	-8.995	-6.372	-4.233	-1.332	5.681
A4	-0.2985	0.8217	0.5852	0.3995	0.1507	-0.511
A5	0.5688	0.6421	0.4329	0.4518	0.4025	-0.03822
A6	-0.2571	-0.4493	-0.2863	-0.3209	-0.2906	0.1354
A7	0.03994	0.1171	0.07647	0.08887	0.08382	-0.04176
A8	-0.001656	-0.009999	-0.006719	-0.007952	-0.007768	0.00387
A9	-31.8	2.918	-5.612	-9.132	-13.37	-34.54
A10	23.25	-2.415	6.56	11.58	16.96	38.72
A11	-3.643	3.949	1.51	-0.0305	-1.535	-7.927
A12	0.3011	-0.4073	-0.1932	-0.04359	0.08915	0.6882

Table 5.5-1 (continued)

Blanketed Fuel – Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) [2] and Cooling Time (Ct) [3]

See notes 1-5 for use of Table 5.5-1

	Fuel Category				
Coefficients	**II-4**	**II-5**	**II-6**	**II-7**	**II-8**
A1	29.37	35.19	23.04	22.79	19.2
A2	-24.3	-29.28	-16.54	-15.59	-11.38
A3	7.817	9.348	5.604	5.376	4.194
A4	-0.7103	-0.8568	-0.512	-0.4954	-0.3908
A5	-0.07327	-0.1466	-0.02031	-0.04904	-0.1143
A6	0.1728	0.2426	0.1193	0.1427	0.1799
A7	-0.05314	-0.07373	-0.03703	-0.04334	-0.04983
A8	0.004935	0.00686	0.00344	0.004006	0.004345
A9	-34.59	-34.09	-14.9	-10.19	-4.624
A10	40.5	41.96	26.55	23.93	19.79
A11	-8.566	-9.168	-4.793	-4.151	-2.908
A12	0.7586	0.8309	0.4338	0.3871	0.2749

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWD/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment". The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ -\ (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3$$

2. Initial enrichment, En, is the nominal central zone U-235 enrichment. Axial blanket material is not considered when determining enrichment. Any enrichment between 2.0 and 5.0 may be used.
3. Cooling time, Ct, is in years. Any cooling time between 72 hours and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.
4. Category I-1 is fresh unburned fuel up to 5.0 wt% U-235 enrichment. No burnup is required. Category I-2 is fresh unburned fuel up to 4.7 wt% U-235 enrichment, fresh unburned fuel up to 5.0 wt% U-235 that contains at least 16 IFBA rods (or an equivalent amount of other burnable absorber), or fuel up to 5.0 wt% U-235 with no burnable absorber rods burned to at least 500 MWD/MTU. For fuel with 16 IFBA rods (or an equivalent amount of other burnable absorber), no burnup is required but any amount of burnup is allowed.
5. This Table applies for any blanketed fuel assembly.

REPLACE TABLE 5.5-2 WITH THE NEXT TWO PAGES

Table 5.5-2

Non-Blanketed Fuel - Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See note 1 for use of Table 5.5-2

Fuel Category	Non-Blanketed Fuel Storage Curve Coefficients[1]							Non-Blanketed Fuel Minimum Burnup[1] (GWd/MTU) for Initial Enrichment[2]					
	A	B	C	D	E	F	G	Cooling Time[3]	1.8 w%	2.5 w%	3.0 w%	3.5 w%	4.0 w%
I-1[4]	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
I-2	18.1371	-0.944126	0.253120	-0.00553408	-0.151450	0.00334051	-29.3574	0	0.23	10.08	16.56	22.56	28.08
								2.5	0.18	9.79	16.08	21.90	27.25
								5	0.14	9.53	15.66	21.33	26.52
								10	0.08	9.11	14.99	20.40	25.34
								15	0.05	8.84	14.55	19.79	24.56
								20	0.03	8.70	14.33	19.48	24.16
II-1	11.9800	0.158287	0.237665	-0.00688305	-0.192273	0.00492032	-14.2029	0	7.87	16.74	23.16	29.67	36.25
								2.5	7.62	16.16	22.36	28.64	35.00
								5	7.38	15.66	21.66	27.75	33.91
								10	6.99	14.85	20.56	26.35	32.22
								15	6.69	14.31	19.85	25.46	31.16
								20	6.49	14.04	19.53	25.10	30.74
II-2	11.8419	0.287918	0.113820	-0.00527641	-0.175033	0.00507248	-9.9305	0	12.32	21.47	28.19	35.04	42.04
								2.5	11.84	20.71	27.22	33.87	40.67
								5	11.41	20.04	26.38	32.86	39.49
								10	10.69	18.98	25.07	31.30	37.68
								15	10.17	18.28	24.25	30.37	36.63
								20	9.83	17.96	23.94	30.06	36.32
II-3	12.6055	0.361578	-0.075193	0.00118870	-0.152297	0.00386780	-8.6212	0	15.24	25.15	32.45	39.93	47.59
								2.5	14.42	24.08	31.20	38.50	45.98
								5	13.70	23.14	30.11	37.25	44.58
								10	12.56	21.68	28.41	35.32	42.41
								15	11.83	20.76	27.35	34.12	41.07
								20	11.51	20.38	26.92	33.65	40.56

Table 5.5-2 (continued)

Non-Blanketed Fuel - Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See note 1 for use of Table 5.5-2

Fuel Category	Non-Blanketed Fuel Storage Curve Coefficients[1]							Non-Blanketed Fuel Minimum Burnup[1] (GWd/MTU) for Initial Enrichment[2]					
	A	B	C	D	E	F	G	Cooling Time[3]	1.8 w%	2.5 w%	3.0 w%	3.5 w%	4.0 w%
II-4	12.6130	0.436168	-0.128105	0.00275389	-0.151579	0.00377707	-7.0392	0	17.08	27.22	34.73	42.45	50.39
								2.5	16.13	26.03	33.36	40.90	48.67
								5	15.31	24.99	32.16	39.56	47.17
								10	14.02	23.37	30.31	37.46	44.83
								15	13.21	22.36	29.15	36.16	43.39
								20	12.88	21.96	28.70	35.67	42.85
II-5	12.6086	0.517311	-0.185177	0.00442008	-0.150482	0.00367344	-5.3438	0	19.03	29.41	37.14	45.12	53.37
								2.5	17.96	28.09	35.64	43.45	51.52
								5	17.02	26.94	34.34	42.00	49.91
								10	15.57	25.16	32.32	39.73	47.41
								15	14.67	24.05	31.06	38.33	45.86
								20	14.32	23.62	30.58	37.80	45.27
II-6	17.1055	-0.116940	0.024104	-0.00410005	-0.262366	0.00761230	-10.7361	0	19.67	31.30	39.53	47.70	55.81
								2.5	18.61	29.81	37.74	45.61	53.42
								5	17.67	28.51	36.18	43.79	51.35
								10	16.15	26.47	33.77	41.01	48.20
								15	15.11	25.18	32.30	39.36	46.36
								20	14.55	24.63	31.76	38.83	45.85
II-7	17.5099	-0.130912	-0.143634	0.00199657	-0.235656	0.00625103	-9.1041	0	21.99	33.85	42.25	50.58	58.84
								2.5	20.65	32.13	40.25	48.31	56.29
								5	19.48	30.63	38.51	46.33	54.08
								10	17.64	28.29	35.82	43.28	50.68
								15	16.45	26.83	34.16	41.42	48.62
								20	15.93	26.25	33.54	40.76	47.92

Table 5.5-2 (continued)

Non-Blanketed Fuel - Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) and Cooling Time (Ct)

See note 1 for use of Table 5.5-2

Fuel Category	Non-Blanketed Fuel Storage Curve Coefficients[1]							Non-Blanketed Fuel Minimum Burnup[1] (GWd/MTU) for Initial Enrichment[2]					
	A	B	C	D	E	F	G	Cooling Time[3]	1.8 w%	2.5 w%	3.0 w%	3.5 w%	4.0 w%
II-8	17.9109	-0.143928	-0.308137	0.00796481	-0.209912	0.00492410	-7.4704	0	24.30	36.41	44.97	53.45	61.87
								2.5	22.69	34.45	42.76	51.01	59.17
								5	21.29	32.75	40.85	48.87	56.82
								10	19.13	30.11	37.86	45.55	53.16
								15	17.80	28.48	36.01	43.48	50.88
								20	17.31	27.86	35.30	42.68	49.98

Notes

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" given in the table for the assembly "cooling time" and "initial enrichment." Alternatively, the specific minimum burnup required for each fuel assembly may be calculated from the following equation: $Bu = A \times En + B \times En^2 + C \times Ct + D \times Ct^2 + E \times Ct \times En + F \times Ct^2 \times En + G$. Only cooling times of 0, 2.5, 5, 10, 15 and 20 years may be used in this equation. Actual cooling time (Ct) is rounded down to the nearest value.

2. Nominal U-235 enrichment.

3. Cooling time in years.

4. Fresh unirradiated fuel up to 4.5 w% U-235 enrichment: No burnup is required.

INSERTS FOR TABLE 5.5-2
(PAGE 1 of 2)

Table 5.5-2

Non-Blanketed Fuel – Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) [2] and Cooling Time (Ct) [3]

See notes 1-4 for use of Table 5.5-2

	Fuel Category					
Coefficients	**I-3**	**II-1**	**II-2a**	**II-2b**	**II-2c**	**II-3**
A1	-35.18	-10.95	-85.44	-41.76	-3.86	-30.33
A2	37.24	7.053	101.1	47.68	2.495	32.89
A3	-12.45	-0.5508	-34.92	-15.55	1.017	-9.444
A4	1.414	0	3.869	1.687	-0.2178	0.9173
A5	1.318	-0.5039	0.262	0.5439	1.056	0.5915
A6	-0.7817	0.3565	-0.2987	-0.6332	-1.204	-0.663
A7	0.13	-0.05129	0.107	0.2377	0.4525	0.2469
A8	-0.00307	0	-0.0111	-0.02719	-0.05352	-0.02849
A9	-21.4	-25.13	42.21	2.733	-7.353	14.57
A10	12.42	18.47	-62.25	-14.23	-1.402	-22.93
A11	-0.3007	-0.9294	27.68	11.17	7.4	14.27
A12	0.008347	0	-3.139	-1.364	-1.051	-1.723

Table 5.5-2 (continued)

Non-Blanketed Fuel – Coefficients to Calculate the Minimum Required Fuel Assembly Burnup (Bu) as a Function of Enrichment (En) [2] and Cooling Time (Ct) [3]

See notes 1-4 for use of Table 5.5-2

	Fuel Category				
Coefficients	**II-4**	**II-5**	**II-6**	**II-7**	**II-8**
A1	-35.55	-22.12	-20.42	-15.32	-82.47
A2	39.21	24.48	21.91	16.33	97.23
A3	-11.72	-6.23	-4.78	-2.554	-33.42
A4	1.18	0.5118	0.2802	-0.00474	3.782
A5	0.7689	0.7824	-0.6527	0.7264	0.4198
A6	-0.8497	-0.8901	0.8281	-0.7902	-0.3959
A7	0.3126	0.3369	-0.2977	0.2894	0.1245
A8	-0.03608	-0.04	0.03405	-0.03319	-0.01113
A9	21.35	18.32	-50.43	22.01	40.38
A10	-29.32	-26.14	56.6	-25.98	-44.56
A11	16.73	16.25	-13.93	15.89	21.97
A12	-2.024	-2.038	1.453	-1.955	-2.568

Notes:

1. All relevant uncertainties are explicitly included in the criticality analysis. For instance, no additional allowance for burnup uncertainty or enrichment uncertainty is required. For a fuel assembly to meet the requirements of a Fuel Category, the assembly burnup must exceed the "minimum burnup" (GWD/MTU) given by the curve fit for the assembly "cooling time" and "initial enrichment". The specific minimum burnup required for each fuel assembly is calculated from the following equation:

$$Bu = (A_1 + A_2{*}En + A_3{*}En^2 + A_4{*}En^3){*}\ exp\ [\ -\ (A_5 + A_6{*}En + A_7{*}En^2 + A_8{*}En^3){*}Ct\ ] + A_9 + A_{10}{*}En + A_{11}{*}En^2 + A_{12}{*}En^3$$

2. Initial enrichment, En, is the nominal U-235 enrichment. Any enrichment between 1.8 and 4.0 may be used.
3. Cooling time, Ct, is in years. Any cooling time between 10 years and 25 years may be used. An assembly with a cooling time greater than 25 years must use 25 years.
4. This Table applies only for pre-EPU fuel assemblies without axial blankets. If an unblanketed assembly is depleted at EPU conditions none of the burnup accrued at EPU conditions can be credited (i.e., only burnup accrued at pre-EPU conditions may be used as burnup credit).

REPLACE TABLE 5.5-3 WITH THE NEXT PAGE

Table 5.5-3

Fuel Categories Ranked by Reactivity[1]

Fuel Category	
Region I	Region II
I-1	II-1
I-2	II-2
	II-3
	II-4
	II-5
	II-6
	II-7
	II-8

Notes

1. Reactivity Rank: Fuel Category is ranked in decreasing order of reactivity, e.g. II-2 is less reactive than II-1, etc.

INSERT FOR TABLE 5.5-3

Table 5.5-3

Fuel Categories Ranked by Reactivity

See notes 1-3 for use of Table 5.5-3

Region I	**I-1**	High Reactivity
	I-2	
	I-3	
Region II	**II-1**	
	II-2a	
	II-2b	
	II-2c	
	II-3	
	II-4	
	II-5	
	II-6	
	II-7	
	II-8	Low Reactivity

Notes:

1. Reactivity Rank: Fuel Category is ranked by decreasing order of reactivity, e.g. Category I-2 is less reactive than Category I-1, etc. The more reactive fuel categories require compensatory measures to be placed in Regions I & II of the SFP, i.e., use of water filled cells or Metamic™ inserts.

2. Category I-1 is fresh fuel up to 5.0 wt% U-235. Category I-2 is fresh fuel up to 4.7 wt% U-235 with no burnable absorber rods, fresh fuel up to 5.0 wt% U-235 with at least 16 IFBA rods (or an equivalent amount of other burnable absorber), or fuel up to 5.0 wt% U-235 with no burnable absorber rods burned to at least 500 MWD/MTU. For fuel with 16 IFBA rods (or an equivalent amount of other burnable absorber), no burnup is required but any amount of burnup is allowed.

3. All Categories except I-1 and I-2 are determined from Tables 5.5-1 and 5.5-2.

REPLACE FIGURE 5.5-1
WITH THE NEXT PAGE

FIGURE 5.5-1

ALLOWABLE REGION I STORAGE ARRAYS

DEFINITION[1,4]

ILLUSTRATION[1,2,3,4]

Array I-A
Checkerboard pattern of Category I-1 assemblies and empty (water filled) cells.

I-1	E
E	I-1

Array I-B
Category I-2 assembly in every cell.

I-2	I-2
I-2	I-2

Array I-C
Category I-1 assemblies and Category I-2 assemblies: Each Category I-1 assembly shall have a full length RCCA in the assembly. The number of Category I-1 assemblies with RCCAs in the assemblies is unrestricted.

I-1	I-2
I-2	I-2

I-1	I-1
I-2	I-2

I-1	I-1
I-2	I-1

I-1	I-1
I-1	I-1

Notes:

1. Fuel Categories are determined from Tables 5.5-1 and 5.5-2.
2. Shaded cells indicate the fuel assembly contains a full length RCCA.
3. **E** indicates an empty (water filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.

INSERT FOR FIGURE 5.5-1

FIGURE 5.5-1

ALLOWABLE REGION I STORAGE ARRAYS

See notes 1-4 for use of Figure 5.5-1

DEFINITION | **ILLUSTRATION**

Array I-A

Checkerboard pattern of Category I-1 assemblies and empty (water-filled) cells.

I-1	X
X	I-1

Array I-B

Category I-3 assembly in every cell.

I-3	I-3
I-3	I-3

Array I-C

Category I-2 and I-3 assemblies. Each Category I-2 shall have a full length RCCA in the assembly.

I-2	I-3
I-3	I-3

I-2	I-2
I-3	I-3

I-2	I-2
I-2	I-3

I-2	I-2
I-2	I-2

Notes:

1. Category I-1 is fresh fuel enriched to 5.0 wt%. I-2 is fresh fuel enriched to 4.7 wt%, fresh fuel enriched to 5.0 wt% with at least 16 IFBA rods (or an equivalent amount of other burnable absorber), or fuel up to 5.0 wt% with no burnable absorber rods burned to at least 500 MWD/MTU. Category I-3 is determined from Tables 5.5-1 and 5.5-2. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Shaded cells indicate that the fuel assembly contains a full length RCCA.
3. **X** indicates an empty (water-filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.

REPLACE FIGURE 5.5-2
WITH THE NEXT TWO PAGES

FIGURE 5.5-2

ALLOWABLE REGION II STORAGE ARRAYS

DEFINITION[1,4]	ILLUSTRATION[1,2,3,4]
Array II-A Category II-1 assembly in three of every four cells: One of every four cells is empty (water-filled).	II-1 II-1 II-1 E
Array II-B Category II-2 assembly in every cell: Two of every four cells contain a Metamic insert (or full length RCCA in the assembly).	II-2 II-2 II-2 II-2 II-2 II-2 II-2 II-2 II-2 II-2 II-2 II-2
Array II-C Checkerboard pattern of Category II-3 and II-5 assemblies: One of every four cells contains a Metamic insert (or full length RCCA in the assembly). Metamic inserts (or RCCAs) may be in either II-3 or II-5 cells.	II-3 II-5 II-5 II-3 II-5 II-3 II-3 II-5
Array II-D Category II-4 assembly in every cell: One of every four cells contains a Metamic insert (or full length RCCA in the assembly).	II-4 II-4 II-4 II-4
Array II-E Checkerboard pattern of Category II-6 and II-8 assemblies.	II-8 II-6 II-6 II-8
Array II-F Category II-7 assembly in every cell.	II-7 II-7 II-7 II-7

Notes:

1. Fuel Categories are determined from Tables 5.5-1 and 5.5-2.
2. Shaded cells indicate either a Metamic insert in the cell or the fuel assembly contains a full length RCCA.
3. E indicates an empty (water filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.

INSERTS FOR FIGURE 5.5-2
(PAGE 1 of 2)

FIGURE 5.5-2

ALLOWABLE REGION II STORAGE ARRAYS

See notes 1-4 for use of Figure 5.5-2

DEFINITION | **ILLUSTRATION**

Array II-A

Category II-1 assemblies in three of every four cells; One of every four cells is empty (water-filled).

II-1	II-1
X	II-1

X	II-1
II-1	II-1

Array II-B1

Checkerboard pattern of Category II-1 and II-3 assemblies with two of every four cells containing a Metamic insert or full length RCCA.

II-1	II-3
II-3	II-1

II-1	II-3
II-3	II-1

II-1	II-3
II-3	II-1

Array II-B2

Category II-2b assembly in every cell with two of every four cells containing a Metamic insert or full length RCCA.

II-2b	II-2b
II-2b	II-2b

II-2b	II-2b
II-2b	II-2b

II-2b	II-2b
II-2b	II-2b

Array II-B3

Checkerboard pattern of Category II-2a and II-2c assemblies with two of every four cells containing a Metamic insert or full length RCCA.

II-2a	II-2c
II-2c	II-2a

II-2a	II-2c
II-2c	II-2a

II-2a	II-2c
II-2c	II-2a

Notes:

1. Fuel categories are determined from Tables 5.5-1 and 5.5-2. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
3. **X** indicates an empty (water-filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.

FIGURE 5.5-2 (continued)

ALLOWABLE REGION II STORAGE ARRAYS

See notes 1-4 for use of Figure 5.5-2

DEFINITION | **ILLUSTRATION**

Array II-C

Checkerboard pattern of Category II-3 and II-5 assemblies with one of every four cells containing a Metamic insert or full length RCCA.

II-3	II-5
II-5	II-3

II-5	II-3
II-3	II-5

Array II-D

Category II-4 assembly in every cell with one out of every four cells containing a Metamic insert or full length RCCA.

II-4	II-4
II-4	II-4

Array II-E

Checkboard pattern of Category II-6 and II-8 assemblies.

II-6	II-8
II-8	II-6

Array II-F

Category II-7 assembly in every cell.

II-7	II-7
II-7	II-7

Notes:

1. Fuel categories are determined from Tables 5.5-1 and 5.5-2. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
3. **X** indicates an empty (water-filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.

REPLACE FIGURE 5.5-3
WITH THE NEXT PAGE

FIGURE 5.5-3

ALLOWABLE INTERFACES BETWEEN REGION II – REGION I ARRAYS

DEFINITION[1,4]

ILLUSTRATION[1,2,3,4,5]

For Array II-A, the empty cell shall be in the row adjacent to the Region I Rack.

Region I Rack

I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2
II-1	**E**	II-1	**E**
II-1	II-1	II-1	II-1

Array II-A

For Array II-B, the reactivity rank of assemblies adjacent to the Region I rack shall be reduced from a rank of II-2 to a reactivity rank of II-4 or lower. The Array II-B pattern shall have the required Metamic insert (or full length RCCA in the assembly) placed in the row adjacent to the Region I rack.

Region I Rack

I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2
II-4	II-4	II-4	II-4
II-2	II-2	II-2	II-2

Array II-B

Region I Rack

I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2
II-4	II-4	II-4	II-4
II-2	II-2	II-2	II-2

Array II-B

Region I Rack

I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2
II-4	II-4	II-4	II-4
II-2	II-2	II-2	II-2

Array II-B

For Arrays II-C and II-D, the Metamic insert (or full length RCCA in the assembly) shall be placed in the row adjacent to the Region I rack.

Region I Rack

I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Array II-C

Region I Rack

I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2
II-5	II-3	II-5	II-3
II-3	II-5	II-3	II-5

Array II-C

Region I Rack

I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-D

Notes:

1. Fuel Categories are determined from Tables 5.5-1 and 5.5-2.
2. Shaded cells indicate either a Metamic insert in the cell or the fuel assembly contains a full length RCCA.
3. **E** indicates an empty (water filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
5. Region I Array I-2 is depicted as the example; however, any Region I array is equally representative.

INSERT FOR FIGURE 5.5-3

FIGURE 5.5-3

INTERFACE RESTRICTIONS BETWEEN REGION II AND REGION I ARRAYS

ALLOWABLE

See notes 1-5 for use of Figure 5.5-3

DEFINITION **ILLUSTRATION**

For Array II-A, the empty cell shall be in the row adjacent to the Region I rack.

Region I Rack

I-3	I-3	I-3	I-3
I-3	I-3	I-3	I-3
II-1	X	II-1	X
II-1	II-1	II-1	II-1

Array II-A

For Arrays requiring two inserts, there shall be an insert in at least every other cell in the row facing the Region I rack.

Region I Rack

I-3	I-3	I-3	I-3
I-3	I-3	I-3	I-3
II-1	II-3	II-1	II-3
II-3	II-1	II-3	II-1

Array II-B1

Region I Rack

I-3	I-3	I-3	I-3
I-3	I-3	I-3	I-3
II-2b	II-2b	II-2b	II-2b
II-2b	II-2b	II-2b	II-2b

Array II-B2

Region I Rack

I-3	I-3	I-3	I-3
I-3	I-3	I-3	I-3
II-2a	II-2c	II-2a	II-2c
II-2c	II-2a	II-2c	II-2a

Array II-B3

For Arrays requiring one insert, the insert shall be placed in the row facing the Region I rack.

Region I Rack

I-3	I-3	I-3	I-3
I-3	I-3	I-3	I-3
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Array II-C

Region I Rack

I-3	I-3	I-3	I-3
I-3	I-3	I-3	I-3
II-5	II-3	II-5	II-3
II-3	II-5	II-3	II-5

Array II-C

Region I Rack

I-3	I-3	I-3	I-3
I-3	I-3	I-3	I-3
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-D

Notes:

1. Fuel categories are determined from Tables 5.5-1 and 5.5-2. In all arrays, an assembly of lower reactivity can replace an assembly of higher reactivity.
2. Shaded cells indicate that the cell contains a Metamic insert or the fuel assembly contains a full length RCCA.
3. **X** indicates an empty (water-filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only. Region I Array I-3 is depicted as the example; however, any Region I array is allowed.
5. Figure 5.5-3 is not applicable to the Region II – cask area rack interface.

REPLACE FIGURE 5.5-4
WITH THE NEXT PAGE

FIGURE 5.5-4

ALLOWABLE REGION II STORAGE ADJACENT TO SPENT FUEL PIT WALLS

DEFINITION[1,4]

ILLUSTRATION[1,2,3,4]

An assembly of rank II-2 placed in the peripheral row of a Region II storage rack shall not be adjacent to a Region I storage rack.

Spent Fuel Pit Wall

II-2	II-2	II-2	II-3
II-3	II-5	II-3	II-5
II-5	II-3	II-5	II-3

Region 2 (Array II-C)

I-2	I-2	I-2	I-2	I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2	I-2	I-2	I-2	I-2
I-2	I-2	I-2	I-2	I-2	I-2	I-2	I-2

Region I (Array I-B)

For Array II-A, the empty cell in the 2x2 II-A array shall be adjacent to the peripheral row that contains the category II-2 assembly(s). For Array II-A only, the peripheral row may contain category II-1 assemblies as the outer two rows will comply with Array II-A requirements.

Spent Fuel Pit Wall

II-2	II-2	II-2	II-2
II-1	E	II-1	E
II-1	II-1	II-1	II-1

Array II-A

II-1	II-1	II-1	II-1
II-1	E	II-1	E
II-1	II-1	II-1	II-1

Array II-A

For Array II-B, the Metamic insert (or full length RCCA in the assembly) shall be adjacent to the peripheral row that contains the category II-2 assembly(s).

Spent Fuel Pit Wall

II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2

Array II-B

II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2

Array II-B

II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2
II-2	II-2	II-2	II-2

Array II-B

For Arrays II-C and II-D, the Metamic insert (or full length RCCA in the assembly) shall be adjacent to the peripheral row that contains the category II-2 assembly(s).

Spent Fuel Pit Wall

II-2	II-2	II-2	II-2
II-5	II-3	II-5	II-3
II-3	II-5	II-3	II-5

Array II-C

II-2	II-2	II-2	II-2
II-5	II-3	II-5	II-3
II-3	II-5	II-3	II-5

Array II-C

II-2	II-2	II-2	II-2
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-D

Notes:

1. Fuel Categories are determined from Tables 5.5-1 and 5.5-2.
2. Shaded cells indicate either a Metamic insert in the cell or the fuel assembly contains a full length RCCA.
3. E indicates an empty (water filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.

INSERT FOR FIGURE 5.5-4

FIGURE 5.5-4

ALLOWABLE EXCEPTIONS TO REGION II STORAGE ARRAYS WHEN ADJACENT TO SPENT FUEL PIT WALLS

See notes 1-6 for use of Figure 5.5-4

DEFINITION **ILLUSTRATION**

For Arrays requiring two inserts, there shall be an insert in at least every other cell adjacent to the peripheral row containing the II-2c assemblies.

Spent Fuel Pit Wall

II-2c	II-2c	II-2c	II-2c
II-3	II-1	II-3	II-1
II-1	II-3	II-1	II-3

Array II-B1

II-2c	II-2c	II-2c	II-2c
II-2b	II-2b	II-2b	II-2b
II-2b	II-2b	II-2b	II-2b

Array II-B2

II-2c	II-2c	II-2c	II-2c
II-2c	II-2a	II-2c	II-2a
II-2a	II-2c	II-2a	II-2c

Array II-B3

For Arrays requiring one insert, the insert shall be adjacent to the peripheral row containing the II-2b assemblies.

Spent Fuel Pit Wall

II-2b	II-2b	II-2b	II-2b
II-5	II-3	II-5	II-3
II-3	II-5	II-3	II-5

Array II-C

II-2b	II-2b	II-2b	II-2b
II-5	II-3	II-5	II-3
II-3	II-5	II-3	II-5

Array II-C

II-2b	II-2b	II-2b	II-2b
II-4	II-4	II-4	II-4
II-4	II-4	II-4	II-4

Array II-D

Notes:

1. Fuel categories are determined from Tables 5.5-1 and 5.5-2. Fuel category of rank II-2b or lower reactivity can be placed in the peripheral row next to the spent fuel pit wall without inserts subject to the constraints listed here. For arrays requiring 2 inserts, the peripheral row must contain category II-2c or lower reactivity. Alternatively, the peripheral row may contain inserts as required for any 2x2 array.
2. Shaded cells indicate either a Metamic insert in the cell or the fuel assembly contains a full length RCCA.
3. **X** indicates an empty (water-filled) cell.
4. Attributes for each 2x2 array are as stated in the definition. Diagram is for illustrative purposes only.
5. There are no restrictions for placement of Arrays II-E and II-F.
6. Any defined Region II array may be placed against the spent fuel pit wall or one of the additional configurations shown above may be used.

5.6 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.6.1 The components identified in Table 5.6-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.6-1.

TABLE 5.6-1

COMPONENT CYCLIC OR TRANSIENT LIMITS

COMPONENT	CYCLIC OR TRANSIENT LIMIT	DESIGN CYCLE OR TRANSIENT
Reactor Coolant System	200 heatup cycles at ≤ 100°F/h and 200 cooldown cycles at ≤ 100°F/h.	Heatup cycle - T_{avg} from ≤ 200°F to ≥ 550°F. Cooldown cycle - T_{avg} from ≥ [illegible] to ≤ 200°F.
	200 pressurizer cooldown cycles at ≤ 200°F/h.	Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	≥ 15% of RATED THERMAL POWER to 0% of RATED THERMAL POWER.
	40 cycles of loss-of-offsite A.C. electrical power.	Loss-of-offsite A.C. electrical ESF Electrical System.
	80 cycles of loss of flow in one reactor coolant loop.	Loss of only one reactor coolant pump.
	400 Reactor trip cycles.	100% to 0% of RATED THERMAL POWER.
	150 ~~leak~~ tests.	Pressurized to ~~≥~~ 2435 psig.
	5 hydrostatic pressure tests.	Pressurized to ~~≥ 3100 psig~~.
Secondary Coolant System	~~6 loss of secondary pressure~~	~~Loss of Secondary pressure~~
	50 ~~leak~~ tests	Pressurized to ≥ 1085 psig
	~~35~~ hydrostatic pressure tests.	Pressurized to ≥ 1356 psig.

200 pressurizer cooldown cycles ≤ 200°F/h from 400 psia

from nominal pressure

Pressurizer cooldown cycle temperatures from ≥ 650°F to ≤ 200°F.

10 cycles of inadvertent auxiliary spray

primary to secondary side

Spray water temperature differential to 560°F

15 primary to secondary side leak tests

Pressurized to 2250 psig

2485 psig and 400°F

hydrostatic pressure

10

15 secondary to primary side leak tests

Pressurized to 840 psig

PROCEDURES AND PROGRAMS (Continued)

9. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionclides in particulate form with half lives greater than 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I;

10. Limitations on the annual dose or dose commitment to any member of the public, beyond the site boundary, due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of Specifications 4.0.2 and 4.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

g. Deleted

h. Containment Leakage Rate Testing Program

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, and as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995, as modified by the following deviations or exemptions:

1) Type A tests will be performed either in accordance with Bechtel Topical Report BN-TOP-1, Revision 1, dated November 1, 1972, or the guidelines of Regulatory Guide 1.163.

2) Type A testing frequency in accordance with NEI 94-01, Revision 0, Section 9.2.3, except:

   a) For Unit 3, the first Type A test performed after the November 1992 Type A test shall be performed no later than November 2007.

   b) For Unit 4, the first Type A test performed after October 1991 shall be performed no later than October 2006.

3) A vacuum test will be performed in lieu of a pressure test for airlock door seals at the required intervals (Amendment Nos. 73 and 77, issued by NRC November 11, 1981).

The peak calculated containment ~~interval~~ internal pressure for the design basis loss of coolant accident, P_a, is ~~49.9 psig.~~ defined here as the containment design pressure of 55 psig.

The maximum allowable containment leakage rate, L_a, at P_a, shall be 0.25% of containment air weight per day.

Leakage Rate acceptance criteria are:

1) The As-found containment leakage rate acceptance criterion is $\leq$ 1.0 L_a. Prior to increasing primary coolant temperature above 200°F following testing in accordance with this program or restoration from exceeding 1.0 L_a, the As-left leakage rate acceptance criterion is $\leq$0.75 L_a, for Type A test.

2) The combined leakage rate for all penetrations subject to Type B or Type C testing is as follows:

PEAKING FACTOR LIMIT REPORT

6.9.1.6 The W(Z) function(s) for Base-Load Operation corresponding to a ± 2% band about the target flux difference and/or a ± 3% band about the target flux difference, the Load-Follow function $F_z(Z)$ and the augmented surveillance turnon power fraction P_T shall be provided to the U.S. Nuclear Regulatory Commission, whenever P_T is <1.0. In the event, the option of Baseload Operation (as defined in Section 4.2.2.3) will not be exercised, the submission of the W(Z) function is not required. Should these values (i.e., W(Z), $F_z(Z)$ and P_T) change requiring a new submittal or an amended submittal to the Peaking 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 Specification 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. All Rods Out position for Specification 3.1.3.2.
5. 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,~~ WCAP-10216-P-A, Revision 1A, " RELAXATION OF CONSTANT AXIAL OFFSET CONTROL F_Q SURVEILLANCE TECHNICAL SPECIFICATION," ~~June 1983.~~ February 1994
2. WCAP-8385, "POWER DISTRIBUTION CONTROL AND LOAD FOLLOWING PROCEDURES - TOPICAL REPORT," September 1974.

The analytical methods used to determine F_Q (Z), $F_{\Delta}H$ and 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-10054-P-A, (proprietary), "Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code," August 1985.

WCAP-16009-P-A, "Realistic Large-Break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", January 2005.

ADM WCAP-10054-P-A Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection into the Broken Loop and COSI Condensation Model," July 1997.

3. ~~WCAP-10054-P, Addendum 2, Revision 1 (proprietary), "Addendum to the Westinghouse Small Break ECCS Evaluation Model Using the NOTRUMP Code: Safety Injection in the Broken Loop and Improved Condensation Model," October 1995.*~~

4. ~~WCAP-12945-P, "Westinghouse Code Qualification Document For Best Estimate LOCA Analysis," Volumes I-V, June 1996.**~~

5. USNRC Safety Evaluation Report, Letter from R. C. Jones (USNRC) to N. J. Liparulo (W), "Acceptance for Referencing of the Topical Report WCAP-12945(P) 'Westinghouse Code Qualification Document for Best Estimate Loss of Coolant Analysis,' " June 28, 1996.**

6. Letter dated June 13, 1996, from N. J. Liparulo (W) to Frank R. Orr (USNRC), "Re-Analysis Work Plans Using Final Best Estimate Methodology."**

7. WCAP-12610-P-A, "VANTAGE+ Fuel Assembly Reference Core Report," S. L. Davidson and T. L. Ryan, April 1995.

The analytical methods used to determine Rod Bank Insertion Limits and the All Rods Out position 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."

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 DPR-41, Florida Power & Light Company Turkey Point Units 3 and 4, Docket Nos. 50-250 and 50-251.

The AFD, $F_Q(Z)$, $F_{\Delta}H$, 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 Resident Inspector, unless otherwise approved by the Commission.

STEAM GENERATOR TUBE INSPECTION REPORT

6.9.1.8 A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with Specification 6.8.4.j, Steam Generator (SG) Program. The report shall include:

a. The scope of inspections performed on each SG,

b. Active degradation mechanisms found,

~~*This reference is only to be used subsequent to NRC approval.~~
**As evaluated in NRC Safety Evaluation dated December 20, 1997.