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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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DEFINITIONS

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 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are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.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.

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TABLE 2.2-1

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
1. Manual Reactor Trip	N.A.	N.A.
2. Power Range, Neutron Flux		
a. High Setpoint	$\leq 108.6\%$ of RTP**	108.0% of RTP**
b. Low Setpoint	$\leq 28.0\%$ of RTP**	$\leq 25\%$ of RTP**
3. Intermediate Range, Neutron Flux	$\leq 31.0\%$ of RTP**	$\leq 25\%$ of RTP**
4. Source Range, Neutron Flux	$\leq 1.4 \times 10^5$ cps	$\leq 10^5$ cps
5. Overtemperature ΔT	See Note 2	See Note 1
6. Overpower ΔT	See Note 4	See Note 3
7. Pressurizer Pressure-Low	≥ 1817 psig	≥ 1835 psig
8. Pressurizer Pressure-High	≤ 2403 psig	≤ 2385 psig
9. Pressurizer Water Level-High	$\leq 92.2\%$ of instrument span	$\leq 92\%$ of instrument span
10. Reactor Coolant Flow-Low	$\geq 89.6\%$ of loop design flow*	90% of loop design flow*
11. Steam Generator Water Level Low-Low	$\geq 15.5\%$ of narrow range instrument span	16% of narrow range instrument span

* Loop design flow = 86,900 gpm

** RTP = Rated Thermal Power

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
12. Steam/Feedwater Flow Mismatch Coincident with Steam Generator Water Level-Low	Feed Flow \leq 20.7% below rated Steam Flow \geq 15.5% of narrow range instrument span	Feed Flow 20% below rated Steam Flow 16% of narrow range instrument span
13. Undervoltage – 4.16 kV Busses A and B	\geq 69% bus voltage	\geq 70% bus voltage
14. Underfrequency – Trip of Reactor Coolant Pump Breaker(s) Open	\geq 55.9 Hz	\geq 56.1 Hz
15. Turbine Trip		
a. Emergency Trip Header Pressure	\geq 901 psig	1000 psig
b. Turbine Stop Valve Closure	Fully Closed***	Fully Closed***
16. Safety Injection Input from ESF	N.A.	N.A.
17. Reactor Trip System Interlocks		
a. Intermediate Range Neutron Flux, P-6	$\geq 6.0 \times 10^{-11}$ amps	Nominal 1×10^{-10} amps

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

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
b. Low Power Reactor Trips Block, P-7		
1) P-10 input	$\leq 13.0\%$ RTP**	Nominal 10% of RTP**
2) Turbine Inlet Pressure	$\leq 13.0\%$ Turbine Power	Nominal 10% Turbine Power
c. Power Range Neutron Flux, P-8	$\leq 48.0\%$ RTP**	Nominal 45% of RTP**
d. Power Range Neutron Flux, P-10	$\geq 7.0\%$ RTP**	Nominal 10% of RTP**
18. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.
19. Reactor Trip Breakers	N.A.	N.A.
20. Automatic Trip and Interlock Logic	N.A.	N.A.

** RTP = RATED THERMAL POWER

TABLE 2.2-1 (Continued)
TABLE NOTATIONS

NOTE 1: OVERTEMPERATURE ΔT (Those values denoted with [*] are specified in the COLR.)

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

Where: ΔT = Measured ΔT by RTD Instrumentation

$\frac{1 + \tau_1 S}{1 + \tau_2 S}$ = Lead/Lag compensator on measured ΔT ; $\tau_1 = [^*]s$, $\tau_2 = [^*]s$

$\frac{1}{1 + \tau_3 S}$ = Lag compensator on measured ΔT ; $\tau_3 = [^*]s$

ΔT_0 = Indicated ΔT at RATED THERMAL POWER

K_1 = [*];

K_2 = [*]/°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} , $\tau_4 = [^*]s$, $\tau_5 = [^*]s$;

T = Average temperature, °F;

$\frac{1}{1 + \tau_6 S}$ = Lag compensator on measured T_{avg} ; $\tau_6 = [^*]s$

T' ≤ [*]°F (Indicated Loop T_{avg} at RATED THERMAL POWER);

K_3 = [*]/psi;

P = Pressurizer pressure, psig;

TABLE 2.2-1 (Continued)TABLE NOTATIONS (Continued)

NOTE 1: (Continued)

$$P' \geq [\star] \text{ psig (Nominal RCS operating pressure);}$$

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

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 $-[\star]\%$ and $+\star\%$, $f_1(\Delta I) = 0$, 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 $-[\star]\%$, the ΔT Trip Setpoint shall be automatically reduced by $[\star]\%$ of its value at RATED THERMAL POWER; and
- (3) For each percent that the magnitude of $q_t - q_b$ exceeds $+\star\%$, the ΔT Trip Setpoint shall be automatically reduced by $[\star]\%$ of its value at RATED THERMAL POWER.

NOTE 2: 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 $f(\Delta I)$ channel. No separate Allowable Value is provided for T_{avg} because this function is part of the ΔT value.

TABLE 2.2-1 (Continued)

TABLE NOTATIONS (Continued)

NOTE 3: OVERPOWER ΔT (Those values denoted with [*] are specified in the COLR.)

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

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 = [*],

K_6 \geq [*]/°F for increasing average temperature and [*]/°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} , $\tau_7 \geq$ [*] 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	=	$[^{\circ}\text{F}]$ for $T > T''$
	=	$[^{\circ}\text{F}]$ for $T \leq T''$,
T	=	As defined in Note 1,
T''	\leq	$[^{\circ}\text{F}]$ (Indicated Loop T_{avg} at RATED THERMAL POWER)
S	=	As defined in Note 1, and
$f_2(\Delta I)$	=	$[^{\circ}\text{F}]$

NOTE 4: 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 T_{avg} because this function is part of the ΔT value.

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REACTIVITY CONTROL SYSTEMS

SHUTDOWN MARGIN - T_{avg} LESS THAN OR EQUAL TO 200°F

LIMITING CONDITION FOR OPERATION

3.1.1.2 The SHUTDOWN MARGIN shall be within the limit specified in the COLR.

APPLICABILITY: MODE 5.

ACTION:

With the SHUTDOWN MARGIN not within limits, immediately 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 equivalent until the required SHUTDOWN MARGIN is restored.

SURVEILLANCE REQUIREMENTS

4.1.1.2 The SHUTDOWN MARGIN shall be determined to be shall be within the limit specified in the COLR:

- a. Within 1 hour after detection of an inoperable control rod(s) and at least once per 12 hours thereafter while the rod(s) is inoperable. If the inoperable control rod is immovable or untrippable, the SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or untrippable control rod(s); and
- b. At least once per 24 hours by consideration of the following factors:
 - 1) Reactor Coolant System boron concentration,
 - 2) Control rod position,
 - 3) Reactor Coolant System average temperature,
 - 4) Fuel burnup based on gross thermal energy generation,
 - 5) Xenon concentration, and
 - 6) Samarium concentration.

REACTIVITY CONTROL SYSTEMS

MODERATOR TEMPERATURE COEFFICIENT

LIMITING CONDITION FOR OPERATION

3.1.1.3 The moderator temperature coefficient (MTC) shall be within the limits specified in the COLR. The maximum upper limit shall be less positive than or equal to $+5.0 \times 10^{-5} \Delta k/k/^{\circ}F$ for all the rods withdrawn, beginning of cycle life (BOL), for power levels up to 70% RATED THERMAL POWER with a linear ramp to $0 \Delta k/k/^{\circ}F$ at 100 % RATED THERMAL POWER.

APPLICABILITY: Beginning of cycle life (BOL) - MODES 1 and 2* only**.
End of cycle life (EOL) - MODES 1, 2, and 3 only**.

ACTION:

- a. With the MTC more positive than the BOL limit specified in the COLR, 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 the BOL limit specified in the COLR 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.

REACTIVITY CONTROL SYSTEMS

LIMITING CONDITION FOR OPERATION

ACTION: (Continued)

- b. With the MTC more negative than the EOL limit specified in the COLR, 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 specified in the COLR, 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 the 300 ppm surveillance limit specified in the COLR (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 the 300 ppm surveillance limit specified in the COLR, the MTC shall be remeasured, and compared to the EOL MTC limit specified in the COLR, at least once per 14 EFPD during the remainder of the fuel cycle.

REACTIVITY CONTROL SYSTEMS

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 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.

REACTIVITY CONTROL SYSTEMS

FLOW PATHS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.2 The following boron injection flow paths shall be OPERABLE:

- a. The source path from a boric acid storage tank via a boric acid transfer pump to the charging pump suction*, and
- b. At least one of the two source paths from the refueling water storage tank to the charging pump suction; and,
- c. The flow path from the charging pump discharge to the Reactor Coolant System via the regenerative heat exchanger.

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

ACTION:

- a. With no boration source path from a boric acid storage tank OPERABLE,
 - 1. Demonstrate the OPERABILITY of the second source path from the refueling water storage tank to the charging pump suction by verifying the flow path valve alignment; and
 - 2. Restore the boration source path from a boric acid storage tank to OPERABLE status within 70 hours or be in at least HOT STANDBY and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F within the next 8 hours; restore the boration source path from a boric acid storage tank to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.
- b. With only one boration source path OPERABLE or the regenerative heat exchanger flow path to the RCS inoperable, restore the required flow paths to OPERABLE status within 70 hours or be in at least HOT STANDBY and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F within the next 8 hours; restore at least two boration source paths to OPERABLE status within the next 72 hours or be in COLD SHUTDOWN within the next 30 hours.
- c. With the boration source path from a boric acid storage tank and the charging pump discharge path via the regenerative heat exchanger inoperable, within one hour initiate boration to a boron concentration equivalent to the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F and go to COLD SHUTDOWN as soon as possible within the limitations of the boration and pressurizer level control functions of the CVCS.

* The flow required in Specification 3.1.2.2.a above shall be isolated from the other unit from the boric acid transfer pump discharge to the charging pump suction.

REACTIVITY CONTROL SYSTEMS

SURVEILLANCE 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 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.

REACTIVITY CONTROL SYSTEMS

CHARGING PUMPS - OPERATING

LIMITING CONDITION FOR OPERATION

3.1.2.3 At least two charging pumps shall be OPERABLE.

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

ACTION:

With only one charging pump OPERABLE, restore at least two charging pumps to OPERABLE status within 70 hours or be in at least HOT STANDBY and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN at 200°F within 8 hours; restore at least two charging pumps to OPERABLE status within 72 hours or be in COLD SHUTDOWN within the next 30 hours.

SURVEILLANCE REQUIREMENTS

4.1.2.3.1 The required charging pumps shall be demonstrated OPERABLE by testing pursuant to Specification 4.0.5. The provisions of Specification 4.0.4 are not applicable for entry into MODES 3 and 4.

REACTIVITY CONTROL SYSTEMS

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 4.0 wt.% (6993 ppm), and
 - 3) A minimum boric acid tanks room temperature of 62°F.
- b. The refueling water storage tank (RWST) with:
 - 1) A minimum indicated borated water volume of 20,000 gallons,
 - 2) A boron concentration 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.

SURVEILLANCE REQUIREMENTS

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 62°F, when it is the source of borated water.

REACTIVITY CONTROL SYSTEMS

BORATED WATER SOURCES - OPERATING

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 62°F.
- b. The refueling water storage tank (RWST) with:
 - 1) A minimum indicated borated water volume of 320,000 gallons,
 - 2) A boron concentration between 2400 ppm and 2600 ppm.
 - 3) A minimum solution temperature of 39°F, and
 - 4) A maximum solution temperature of 100°F.

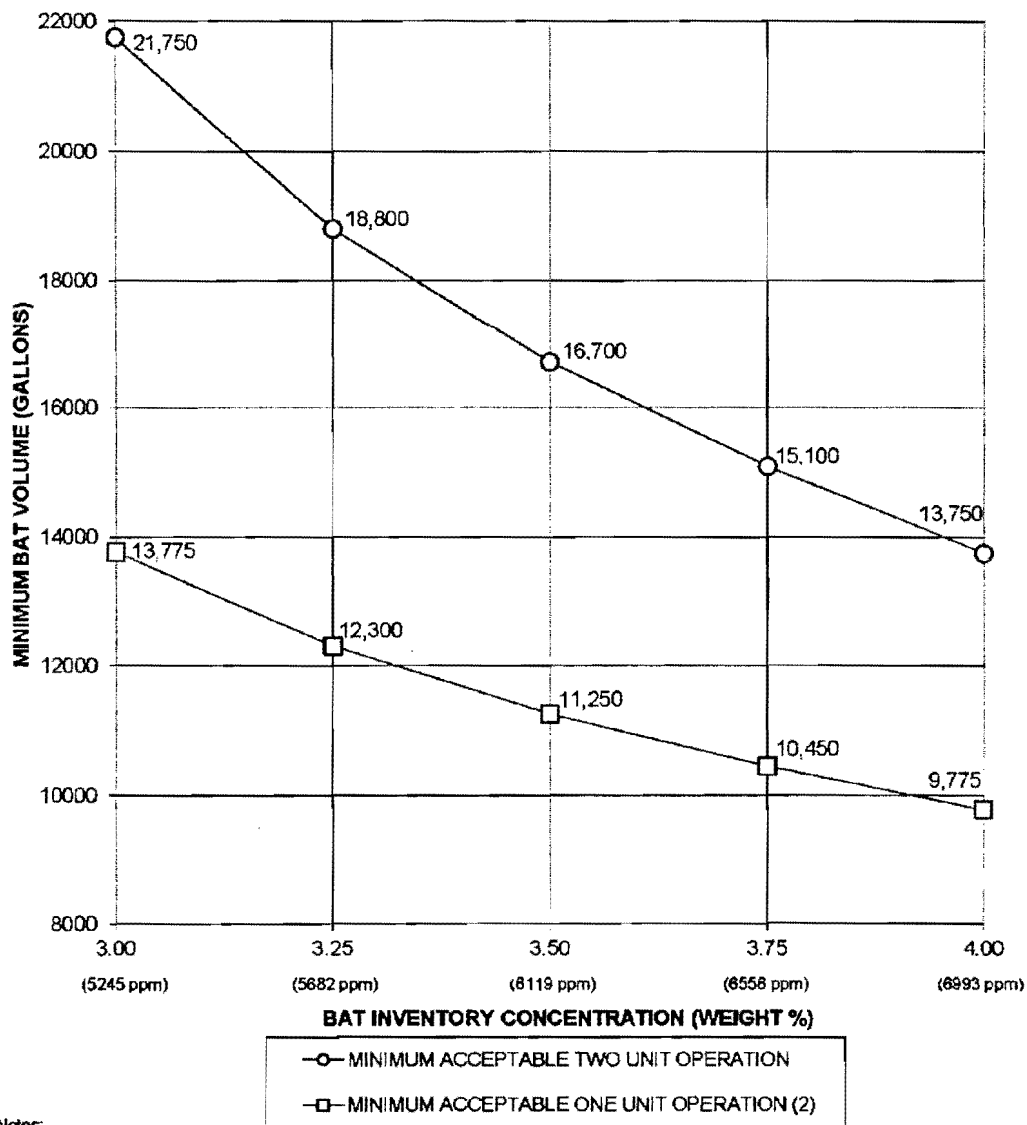
APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

- a. With the required Boric Acid Storage System inoperable verify that the RWST is OPERABLE; restore the system to OPERABLE status within 70 hours or be in at least HOT STANDBY within the next 8 hours* and borated to a boron concentration equivalent to at least the required SHUTDOWN MARGIN at COLD SHUTDOWN 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 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 sixteen hours.

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.

REACTIVITY CONTROL SYSTEMS

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
 - 3) Verifying that the temperature of the boric acid tanks room is greater than or equal to 62°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.

POWER DISTRIBUTION LIMITS

3/4.2.5 DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

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

- a. Reactor Coolant System T_{avg} is less than or equal to the limit specified in the COLR
- b. Pressurizer Pressure is greater than or equal to the limit specified in the COLR*, and
- c. Reactor Coolant System Flow $\geq 270,000$ gpm

APPLICABILITY: MODE 1.

ACTION:

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

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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
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. Emergency Trip Header Pressure	3	2	2	1	12
b. Turbine Stop Valve Closure	2	2	2	1	12

|

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
16. Safety Injection Input from ESF	2	1	2	1, 2	8
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2	1	2	2#	7
b. Low Power Reactor Trips Block, P-7					
P-10 Input	4	2	3	1	7
or					
Turbine Inlet Pressure	2	1	2	1	7
c. Power Range Neutron Flux, P-8	4	2	3	1	7
d. Power Range Neutron Flux, P-10	4	2	3	1, 2	7
18. Reactor Coolant Pump Breaker Position Trip					
a. Above P-8	1/breaker	1	1/breaker	1	11
b. Above P-7 and below P-8	1/breaker	2	1/breaker	1	11
19. Reactor Trip Breakers	2	1	2	1, 2	8, 10
	2	1	2	3*, 4*, 5*	9
20. Automatic Trip and Interlock logic	2	1	2	1, 2	8
	2	1	2	3*, 4*, 5*	9

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
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), ^{(a), (b)} R(4) ^{(a), (b)}	Q ^{(a), (b)}	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 ^{(a), (b)}	Q ^{(a), (b)}	N.A.	N.A.	1, 2
6. Overpower ΔT	S	R ^{(a), (b)}	Q ^{(a), (b)}	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 ^{(a), (b)}	Q ^{(a), (b)}	N.A.	N.A.	1
11. Steam Generator Water Level--Low-Low	S	R ^{(a), (b)}	Q ^{(a), (b)}	N.A.	N.A.	1, 2

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

<u>FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
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. Emergency Trip Header Pressure	N.A.	R ^{(a), (b)}	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 Inlet 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

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.
- (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 UFSAR Section 7.2.
- (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.

TABLE 3.3-2ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
1. Safety Injection					
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

<u>FUNCTIONAL UNIT</u>	<u>TOTAL NO. OF CHANNELS</u>	<u>CHANNELS TO TRIP</u>	<u>MINIMUM CHANNELS OPERABLE</u>	<u>APPLICABLE MODES</u>	<u>ACTION</u>
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 Actua- tion 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, 3	15
6. Auxiliary Feedwater###					
a. Automatic Actua- tion 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		
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	\leq A function defined as follows: A ΔP corresponding to 41.2% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 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

TABLE 3.3-3 (Continued)
ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTS

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
Coincident with: Steam Generator Pressure--Low(4) or T _{avg} --Low	≥607 psig ≥542.5°F	614 psig ≥543°F
2. Containment Spray		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Containment Pressure--High- High Coincident with: Containment Pressure--High	≤22.6 psig ≤4.5 psig	≤20.0 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

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
4. Steam Line Isolation (Continued)		
b. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
c. Containment Pressure--High-High Coincident with: Containment Pressure--High	≤22.6 psig ≤ 4.5 psig	≤20. psig ≤4.0 psig
d. Steam Line Flow--High	≤A function defined as follows: A ΔP corresponding to 41.2% steam flow at 0% load increasing linearly from 20% load to a value corresponding to 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) or T _{avg} -- Low	≥607 psig ≥542.5°F	614 psig ≥543°F
5. 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

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE</u>	<u>TRIP SETPOINT</u>
5. Feedwater Isolation (Continued)		
c. Steam Generator Water Level High-High	≤80.5% of narrow range instrument span	80% of narrow range instrument span
6. Auxiliary Feedwater (3)		
a. Automatic Actuation Logic and Actuation Relays	N.A.	N.A.
b. Steam Generator Water Level—Low-Low	≥15.5% of narrow range instrument span.	16% 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

<u>FUNCTIONAL UNIT</u>	<u>ALLOWABLE VALUE#</u>	<u>TRIP SETPOINT</u>
7. Loss of Power (Continued)		
b. 480V Load Centers Undervoltage		
<u>Load Center</u>		
3A	[]	430V \pm 3V (10 sec \pm 1 sec delay)
3B	[]	438V \pm 3V (10 sec \pm 1 sec delay)
3C	[]	434V \pm 3V (10 sec \pm 1 sec delay)
3D	[]	434V \pm 3V (10 sec \pm 1 sec delay)
4A	[]	435V \pm 3V (10 sec \pm 1 sec delay)
4B	[]	434V \pm 3V (10 sec \pm 1 sec delay)
4C	[]	434V \pm 3V (10 sec \pm 1 sec delay)
4D	[]	430V \pm 3V (10 sec \pm 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

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

TABLE 3.3-3 (Continued)ENGINEERED SAFETY FEATURES ACTUATION SYSTEM
INSTRUMENTATION TRIP SETPOINTSTABLE NOTATIONS

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

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

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

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.

(4) Time constants utilized in lead-lag controller for Steam Generator Pressure-Low and Steam Line Pressure-Low are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds. CHANNEL CALIBRATION shall ensure that these time constants are adjusted to these values.

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

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

<u>CHANNEL FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST #</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
1. Safety Injection						
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: Steam Generator Pressure--Low or Tavg--Low	S	R ^{(a)(b)}	Q(5) ^{(a)(b)}	N.A.	N.A.	1, 2, 3(3)
	S	R ^{(a)(b)}	Q(5) ^{(a)(b)}	N.A.	N.A.	1, 2, 3(3)
	S	R	Q(5)	N.A.	N.A.	1, 2, 3(3)

TABLE 4.3-2 (Continued)

ENGINEERED SAFETY FEATURES ACTUATION SYSTEM INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

<u>CHANNEL FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST #</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
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	N.A.	R	M(1)	1, 2, 3
d. Steam Line Flow--High	S(3)	R ^{(a)(b)}	Q(5) ^{(a)(b)}	N.A.	N.A.	1, 2, 3
Coincident with: Steam Generator Pressure--Low	S(3)	R ^{(a)(b)}	Q(5) ^{(a)(b)}	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, 3
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

<u>CHANNEL FUNCTIONAL UNIT</u>	<u>CHANNEL CHECK</u>	<u>CHANNEL CALIBRATION</u>	<u>ANALOG CHANNEL OPERATIONAL TEST</u>	<u>TRIP ACTUATING DEVICE OPERATIONAL TEST</u>	<u>ACTUATION LOGIC TEST #</u>	<u>MODES FOR WHICH SURVEILLANCE IS REQUIRED</u>
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

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

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 UFSAR Section 7.2
- (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.

TABLE 3.3-4 (Continued)
TABLE NOTATIONS

* During CORE ALTERATIONS or movement of irradiated fuel within the containment comply with Specification 3/4.9.13.

** With irradiated fuel in the spent fuel pits.

Unit 4 Spent Fuel Pool Area is monitored by Plant Vent radioactivity instrumentation.

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

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

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.

ACTION STATEMENTS

ACTION 26 - In MODES 1 thru 4: With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:

- 1) A Containment sump level monitoring system is OPERABLE,
- 2) Appropriate grab samples are obtained and analyzed at least once per 24 hours,
- 3) A Reactor Coolant System water inventory balance is performed at least once per 8*** hours except when operating in shutdown cooling mode, and
- 4) Containment Purge, Exhaust and Instrument Air Bleed Valves are maintained closed. ****

Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours (ACTION 27 applies in MODES 5 and 6).

*** Not required to be performed until 12 hours after establishment of steady state operation.

**** Instrument Air Bleed Valves may be opened intermittently under administrative controls.

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY VALVES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.4.2.1 A minimum of one pressurizer Code safety valve shall be OPERABLE* with a lift setting of 2465 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 $\pm 1\%$ of the lift setting value.

REACTOR COOLANT SYSTEM

OPERATING

LIMITING CONDITION FOR OPERATION

3.4.2.2 All pressurizer Code safety valves shall be OPERABLE with a lift setting of 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 $\pm 1\%$ of the lift setting value.

REACTOR COOLANT SYSTEM

3/4.4.6 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.6.1 The following Reactor Coolant System Leakage Detection Systems shall be OPERABLE:

- a. The Containment Atmosphere Gaseous or Particulate Radioactivity Monitoring System, and
- b. A Containment Sump Level Monitoring System.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

- a. With both the Particulate and Gaseous Radioactivity Monitoring Systems inoperable, operation may continue for up to 7 days provided:
 - 1) A Containment Sump Level Monitoring System is OPERABLE;
 - 2) Appropriate grab samples are obtained and analyzed at least once per 24 hours;
 - 3) A Reactor Coolant System water inventory balance is performed at least once per 8* hours except when operating in shutdown cooling mode; and
 - 4) Containment Purge, Exhaust and Instrument Air Bleed valves are maintained closed.**Otherwise, be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.
- b. With no Containment Sump Level Monitoring System operable, restore at least one Containment Sump Level Monitoring System to OPERABLE status within 7 days, or be in at least HOT STANDBY within 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1 The Leakage Detection System shall be demonstrated OPERABLE by:

- a. Containment Atmosphere Gaseous and Particulate Monitoring System-performance of CHANNEL CHECK, CHANNEL CALIBRATION and ANALOG CHANNEL OPERATIONAL TEST at the frequencies specified in Table 4.3-3, and
- b. Containment Sump Level Monitoring System-performance of CHANNEL CALIBRATION at least once per 18 months.

* Not required to be performed until 12 hours after establishment of steady state operation.

** Instrument Air Bleed valves may be opened intermittently under administrative controls.

MATERIAL PROPERTY BASIS

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

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

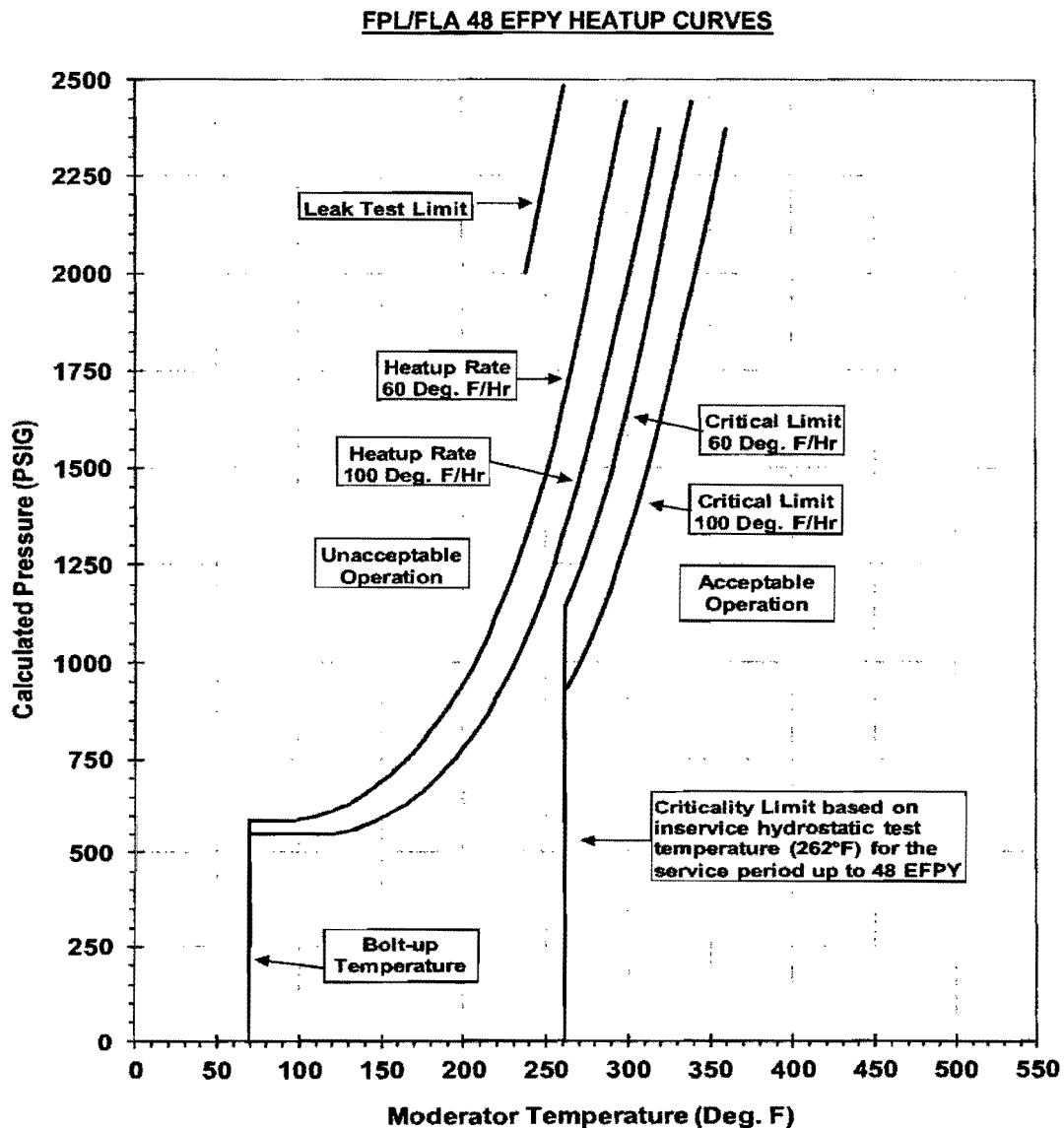


FIGURE 3.4-2 Turkey Point Units 3 and 4 Reactor Coolant System Heatup Limitations (Heatup Rates of 60 and 100°F/hr) Applicable for 48 EFY (Without Margins for Instrumentation Errors)

MATERIAL PROPERTY BASIS

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

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

FPL 48 EFY COOLDOWN CURVES

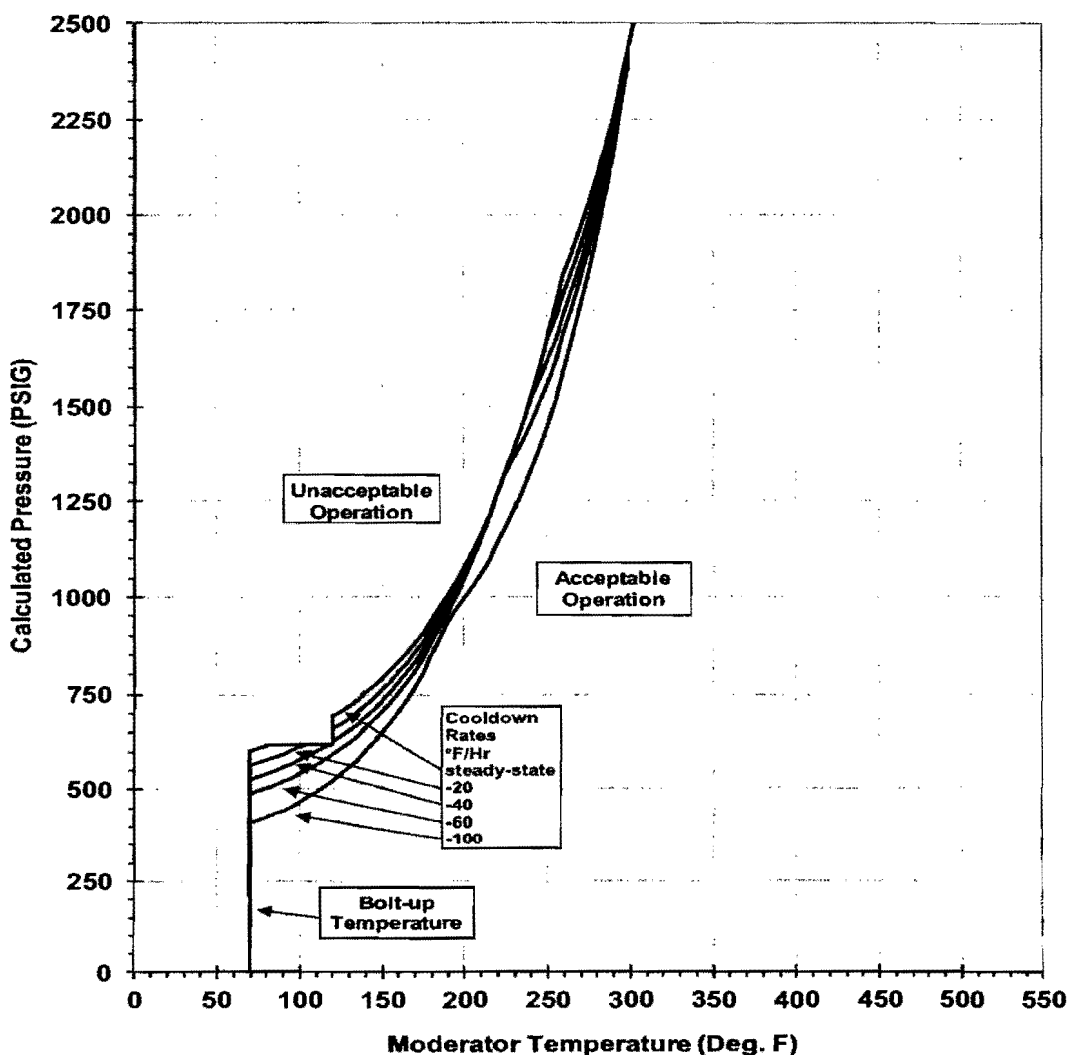


FIGURE 3.4-3 Turkey Point Units 3 and 4 Reactor Coolant System Cooldown Limitations (Cooldown Rates of 0, 20, 40, 60 and 100°F/hr) Applicable for 48 EFY (Without Margins for Instrumentation Errors)

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:

- a. Two power-operated relief valves (PORVs) with a lift setting of ≤ 448 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.

EMERGENCY CORE COOLING SYSTEMS

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 2300 and 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.

EMERGENCY CORE COOLING SYSTEMS

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 boron concentration 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.

CONTAINMENT SYSTEMS

INTERNAL PRESSURE

LIMITING CONDITION FOR OPERATION

3.6.1.4 Primary containment internal pressure shall be maintained between -2 and +1 psig. |

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

ACTION:

With the containment internal pressure outside of the limits above, restore the internal pressure to within the limits within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.4 The primary containment internal pressure shall be determined to be within the limits at least once per 12 hours.

TABLE 3.7-1
MAXIMUM ALLOWABLE POWER LEVEL WITH
INOPERABLE STEAM LINE SAFETY VALVES DURING THREE LOOP OPERATION

<u>MAXIMUM NUMBER OF INOPERABLE SAFETY VALVES ON ANY OPERATING STEAM GENERATOR</u>	<u>MAXIMUM ALLOWABLE POWER LEVEL (PERCENT OF RATED THERMAL POWER)</u>	
1	44	
2	27	
3	10	

TABLE 3.7-2
STEAM LINE SAFETY VALVES PER LOOP

<u>VALVE NUMBER</u>			<u>LIFT SETTING ($\pm 3\%$)* **</u>	<u>ORIFICE SIZE SQUARE INCHES</u>
<u>Loop A</u>	<u>Loop B</u>	<u>Loop C</u>		
1. RV1400	RV1405	RV1410	1085 psig	16
2. RV1401	RV1406	RV1411	1100 psig	16
3. RV1402	RV1407	RV1412	1105 psig	16
4. RV1403	RV1408	RV1413	1105 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 $\pm 1\%$ of the lift setting value listed in Table 3.7-2.

PLANT SYSTEMS

STANDBY FEEDWATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.6 Two Standby Steam Generator Feedwater Pumps shall be OPERABLE* and at least 145,000 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.
- c. With less than 145,000 gallons of water indicated in the Demineralized Water Storage Tank restore the available volume to at least 145,000 gallons indicated within 24 hours or submit a SPECIAL REPORT per 3.7.1.6d.
- d. If a SPECIAL REPORT is required per the above specifications submit a report describing the cause of the inoperability, action taken and a schedule for restoration within 30 days in accordance with 6.9.2.

SURVEILLANCE REQUIREMENTS

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

4.7.1.6.2 At least monthly verify the standby feedwater pumps are OPERABLE by testing in recirculation on a STAGGERED TEST BASIS.

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

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

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

PLANT SYSTEMS

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.

ELECTRICAL POWER SYSTEMS

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 3950-4350 volts and 60 ± 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.

ELECTRICAL POWER SYSTEMS

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 3950-4350 volts and frequency at 60 ± 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.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 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 3950-4350 volts and 60 ± 0.6 Hz during this test.
- 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 3950-4350 volts and 60 ± 0.6 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 3950-4350 volts and 60 ± 0.6 Hz during this test; and
 - 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 3950-4350 volts and 60 ± 0.6 Hz within 15 seconds after the start signal; the steady-state generator voltage and frequency

* 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

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

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 3950-4350 volts and 60 ± 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 $\pm 10\%$ 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 3950-4350 volts and 60 ± 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.

ELECTRICAL POWER SYSTEMS

SURVEILLANCE REQUIREMENTS (Continued)

- 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 ± 0.6 Hz frequency and 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 2300 ppm.

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 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.

* 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.

REFUELING OPERATIONS

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 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 2300 ppm, suspend movement of spent fuel in the Spent Fuel Pit and initiate action to restore boron concentration to 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 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.

DESIGN FEATURES

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 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 cask area storage rack.
- d. A maximum enrichment loading for fuel assemblies of 5.0 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.
- g. The Metamic neutron absorber inserts shall have a minimum certified ^{10}B areal density greater than or equal to 0.015 grams $^{10}\text{B}/\text{cm}^2$.

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 a nominal 4.5 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.

DESIGN FEATURES

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

<u>COMPONENT</u>	<u>CYCLIC OR TRANSIENT LIMIT</u>	<u>DESIGN CYCLE OR TRANSIENT</u>
Reactor Coolant System	200 heatup cycles at $\leq 100^{\circ}\text{F/h}$ and 200 cooldown cycles at $\leq 100^{\circ}\text{F/h}$.	Heatup cycle - T_{avg} from $\leq 200^{\circ}\text{F}$ to $\geq 550^{\circ}\text{F}$. Cooldown cycle - T_{avg} from $\geq 550^{\circ}\text{F}$ to $\leq 200^{\circ}\text{F}$.
	200 pressurizer cooldown cycles at $\leq 200^{\circ}\text{F/h}$ from nominal pressure.	Pressurizer cooldown cycle temperatures from $\geq 650^{\circ}\text{F}$ to $\leq 200^{\circ}\text{F}$.
	200 pressurizer cooldown cycles $\leq 200^{\circ}\text{F/h}$ from 400 psia.	Pressurizer cooldown cycle temperatures from $\geq 650^{\circ}\text{F}$ to $\leq 200^{\circ}\text{F}$.
	80 loss of load cycles, without immediate Turbine or Reactor trip.	$\geq 15\%$ 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.
	10 cycles of inadvertent auxiliary spray	Spray water temperature differential to 560°F .
	150 primary to secondary side leak tests.	Pressurized to 2435 psig.
	15 primary to secondary side leak tests.	Pressurized to 2250 psig.
Secondary Coolant System	5 hydrostatic pressure tests.	Pressurized to 2485 psig and 400°F .
	50 hydrostatic pressure tests	Pressurized to 1085 psig
	10 hydrostatic pressure tests.	Pressurized to 1356 psig.
	15 secondary to primary side leak tests	Pressurized to 840 psig

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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 radionuclides 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 internal pressure for the design basis loss of coolant accident, P_a , is defined here as the containment design pressure of 55 psig.

The maximum allowable containment leakage rate, L_a , at P_a , shall be 0.20% 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:

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3. 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.
4. WCAP-16009-P-A, "Realistic Large-break LOCA Evaluation Methodology Using the Automated Statistical Treatment of Uncertainty Method (ASTRUM)", January 2005.
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 Overtemperature ΔT and Overpower ΔT shall be those previously reviewed and approved by the NRC in:

1. WCAP-8745-P-A, "Design Basis for the Thermal Overtemperature ΔT and Overpower ΔT Trip Functions," September 1986
2. WCAP-9272-P-A, "Westinghouse Reload Safety Evaluation Methodology," July 1985

The analytical methods used to determine Safety Limits, Shutdown Margin - $T_{avg} > 200^\circ\text{F}$, Shutdown Margin - $T_{avg} \leq 200^\circ\text{F}$, Moderator Temperature Coefficient, DNB Parameters, 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, $F_Q(Z)$, $F_{\Delta H}$, $K(Z)$, Safety Limits, Overtemperature ΔT , Overpower ΔT , Shutdown Margin - $T_{avg} > 200^\circ\text{F}$, Shutdown Margin - $T_{avg} \leq 200^\circ\text{F}$, Moderator Temperature Coefficient, DNB Parameters, 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.

**As evaluated in NRC Safety Evaluation dated December 20, 1997.