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1.1 Definitions (continued)

**ENGINEERED SAFETY
FEATURE (ESF) RESPONSE
TIME**

The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured. In lieu of measurement, response time may be verified for selected components provided that the components and methodology for verification have been previously reviewed and approved by the NRC.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except reactor coolant pump (RCP) seal water leakoff), that is captured and conducted to collection systems or a sump or collecting tank;
2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or
3. Reactor Coolant System (RCS) LEAKAGE through a steam generator to the Secondary System (primary to secondary LEAKAGE);

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except primary to secondary LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

(continued)

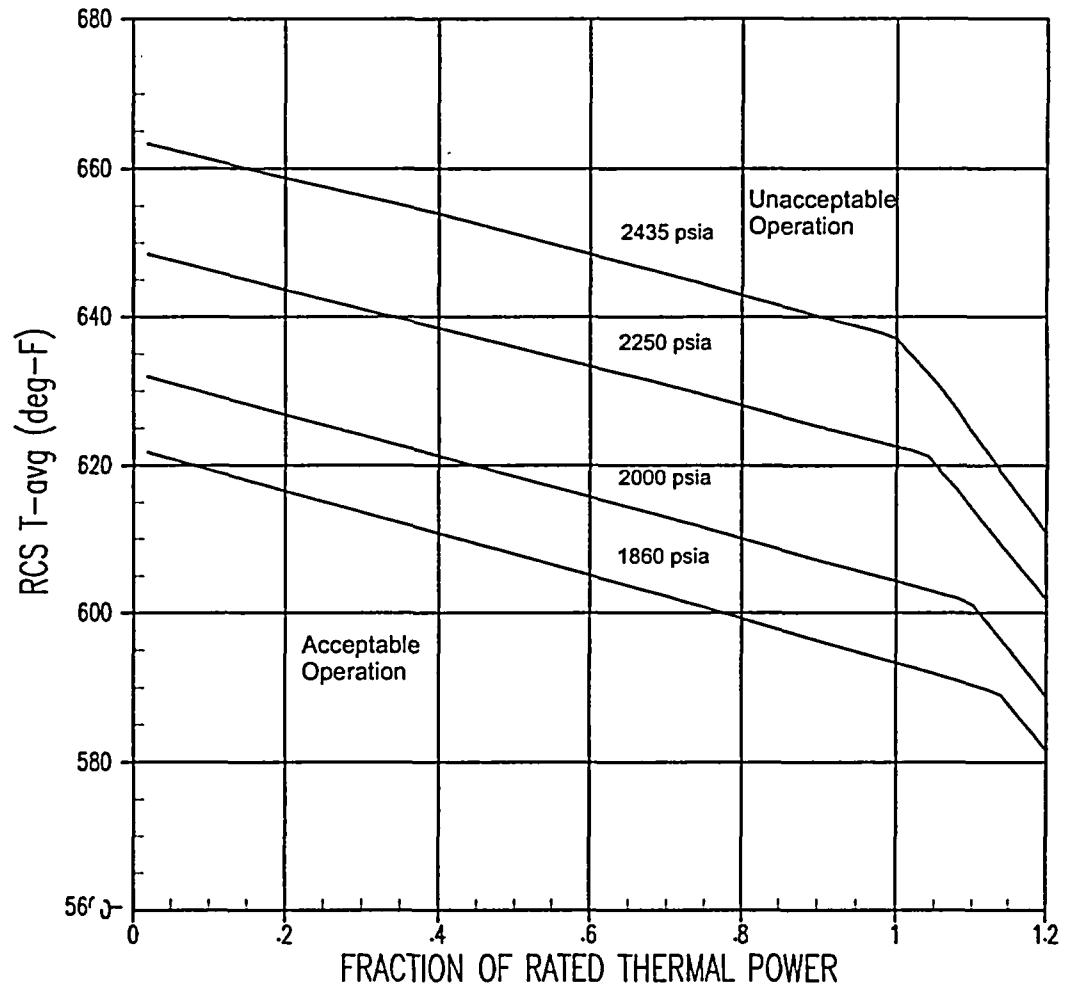


Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
W. Not used.		
X. One or more Containment Pressure - Environmental Allowance Modifier channel(s) inoperable.	X.1 Place channel(s) in trip.	72 hours
	<u>OR</u>	
	X.2 Be in MODE 3.	78 hours

Table 3.3.1-1 (page 1 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.14	NA
	3 ^(b) , 4 ^(b) , 5 ^(b)	2	C	SR 3.3.1.14	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 112.3% RTP
b. Low	1 ^(c) , 2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 SR 3.3.1.16	≤ 28.3% RTP
3. Power Range Neutron Flux Rate - High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.16	≤ 6.3 % RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1 ^(c) , 2 ^(d)	2	F, G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 35.3% RTP

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

(b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.

(c) Below the P-10 (Power Range Neutron Flux) interlock.

(d) Above the P-6 (Intermediate Range Neutron Flux) interlock.

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
5. Source Range Neutron Flux	2 ^(e)	2	I, J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.6 E5 cps
	3 ^(b) , 4 ^(b) , 5 ^(b)	2	J, K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1.6 E5 cps
6. Overtemperature ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 1 (Page 3.3-23)
7. Overpower ΔT	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	Refer to Note 2 (Page 3.3-24)
8. Pressurizer Pressure					
a. Low	1 ^(g)	4	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 1874 psig
b. High	1,2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2393 psig

- (a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
- (b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
- (e) Below the P-6 (Intermediate Range Neutron Flux) interlock.
- (g) Above the P-7 (Low Power Reactor Trips Block) interlock.

Table 3.3.1-1 (page 3 of 8)
Reactor Trip System Instrumentation

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
9.	Pressurizer Water Level - High	1 ^(g)	3	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 93.8% of instrument span
10.	Reactor Coolant Flow - Low	1 ^(g)	3 per loop	M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 88.8% of indicated loop flow
11.	Not Used					
12.	Undervoltage RCPs	1 ^(g)	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 10105 Vac
13.	Underfrequency RCPs	1 ^(g)	2/bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.16	≥ 57.1 Hz
14.	Steam Generator (SG) Water Level Low-Low ^(l)					
a.	Steam Generator Water Level Low-Low (Adverse Containment Environment)	1, 2	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 20.6% ^(q) of Narrow Range Instrument Span

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

(g) Above the P-7 (Low Power Reactor Trips Block) interlock.

(l) The applicable MODES for these channels in Table 3.3.2-1 are more restrictive.

(m) Not used.

(q) 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases.

Table 3.3.1-1 (page 4 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
14. Steam Generator (SG) Water Level Low-Low ^(l)					
b. Steam Generator Water Level Low-Low (Normal Containment Environment)	1 ^(p) , 2 ^(p)	4 per SG	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≥ 16.6% ^(a) of Narrow Range Instrument Span
c. Not used.					
d. Containment Pressure - Environmental Allowance Modifier	1, 2	4	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.16	≤ 2.0 psig
15. Not Used					

- (a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
- (l) The applicable MODES for these channels in Table 3.3.2-1 are more restrictive.
- (n) Not used.
- (o) Not used.
- (p) Except when the Containment Pressure – Environmental Allowance Modifier channels in the same protection sets are tripped.
- (q) 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases.

Table 3.3.1-1 (page 5 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
16. Turbine Trip					
a. Low Fluid Oil Pressure	1 ^(j)	3	O	SR 3.3.1.10 SR 3.3.1.15	≥ 539.42 psig
b. Turbine Stop Valve Closure	1 ^(j)	4	P	SR 3.3.1.10 SR 3.3.1.15	≥ 1% open
17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q	SR 3.3.1.14	NA
18. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2 ^(e)	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6E-11 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.5	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 51.3% RTP
d. Power Range Neutron Flux, P-9	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 53.3% RTP

(a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.

(e) Below the P-6 (Intermediate Range Neutron Flux) interlock.

(j) Above the P-9 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 6 of 8)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
18. Reactor Trip System Interlocks					
e. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	≥ 6.7% RTP and ≤ 12.4% RTP
f. Turbine Impulse Pressure, P-13	1	2	T	SR 3.3.1.10 SR 3.3.1.13	≤ 12.4% turbine power
19. Reactor Trip Breakers (RTBs) ^(k)	1,2 3 ^(b) , 4 ^(b) , 5 ^(b)	2 trains 2 trains	R C	SR 3.3.1.4 SR 3.3.1.4	NA NA
20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms ^(k)	1,2 3 ^(b) , 4 ^(b) , 5 ^(b)	1 each per RTB 1 each per RTB	U C	SR 3.3.1.4 SR 3.3.1.4	NA NA
21. Automatic Trip Logic	1,2 3 ^(b) , 4 ^(b) , 5 ^(b)	2 trains 2 trains	Q C	SR 3.3.1.5 SR 3.3.1.5	NA NA

- (a) The Allowable Value defines the limiting safety system setting except for Trip Functions 14.a and 14.b (the Nominal Trip Setpoint defines the limiting safety system setting for these Trip Functions). See the Bases for the Nominal Trip Setpoints.
- (b) With Rod Control System capable of rod withdrawal or one or more rods not fully inserted.
- (k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Note 1: Overtemperature ΔT

The Overtemperature ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.23% of ΔT span (1.85% RTP).

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

Where: ΔT is measured RCS ΔT , °F.
 ΔT_o is the indicated ΔT at RTP, °F.
 s is the Laplace transform operator, sec⁻¹.
 T is the measured RCS average temperature, °F.
 T' is the nominal T_{avg} at RTP, $\leq 585.3^\circ\text{F}$.

P is the measured pressurizer pressure, psig.
 P' is the nominal RCS operating pressure = 2235 psig.

$K_1 = 1.1950$	$K_2 = 0.0251/^\circ\text{F}$	$K_3 = 0.00116/\text{psig}$
$\tau_1 \geq 8 \text{ sec}$	$\tau_2 \leq 3 \text{ sec}$	$\tau_3 = 0 \text{ sec}$
$\tau_4 \geq 28 \text{ sec}$	$\tau_5 \leq 4 \text{ sec}$	$\tau_6 = 0 \text{ sec}$
$f_1(\Delta I) =$		
-0.0325 {21% + ($q_t - q_b$)}		when $q_t - q_b < -21\% \text{ RTP}$
0% of RTP		when $-21\% \text{ RTP} \leq q_t - q_b \leq 8\% \text{ RTP}$
0.02973 {($q_t - q_b$) - 8%}		when $q_t - q_b > 8\% \text{ RTP}$

where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following setpoint by more than 1.21% of ΔT span (1.82% RTP).

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

Where: ΔT is measured RCS ΔT , °F.
 ΔT_o is the indicated ΔT at RTP, °F.
 s is the Laplace transform operator, sec⁻¹.
 T is the measured RCS average temperature, °F.
 T'' is the nominal T_{avg} at RTP, $\leq 585.3^\circ\text{F}$.

$$K_4 = 1.1073$$

$$K_5 = 0.02/^\circ\text{F for increasing } T_{avg} \\ 0/^\circ\text{F for decreasing } T_{avg}$$

$$K_6 = 0.0015/^\circ\text{F when } T > T'' \\ 0/^\circ\text{F when } T \leq T'' \\ \tau_3 = 0 \text{ sec}$$

$$\tau_1 \geq 8 \text{ sec} \\ \tau_6 = 0 \text{ sec}$$

$$\tau_2 \leq 3 \text{ sec} \\ \tau_7 \geq 10 \text{ sec}$$

$$f_2(\Delta I) = 0\% \text{ RTP for all } \Delta I.$$

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
M. Not used.		
N. One or more Containment Pressure - Environmental Allowance Modifier channel(s) inoperable.	N.1 Place channel(s) in trip.	72 hours
	<u>OR</u>	
	N.2.1 Be in MODE 3.	78 hours
	<u>AND</u>	
	N.2.2 Be in MODE 4.	84 hours
O. One channel inoperable.	O.1 Place channel in trip.	1 hour
	<u>AND</u>	
	O.2 Restore channel to OPERABLE status.	During performance of the next required COT

(continued)

Table 3.3.2-1 (page 1 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
1. Safety Injection					
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.13	NA
c. Containment Pressure - High 1	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 4.5 psig
d. Pressurizer Pressure - Low	1,2,3 ^(b)	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 1834 psig
e. Steam Line Pressure - Low	1,2,3 ^(b)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 610 psig ^{(c) (s)}
2. Containment Spray					
a. Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA

(a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.

(b) Above the P-11 (Pressurizer Pressure) interlock and below P-11 unless the Function is blocked.

(c) Time constants used in the lead/lag controller are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.

(s) 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases.

Table 3.3.2-1 (page 2 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION		APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
2.	Containment Spray					
c.	Containment Pressure High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 28.3 psig
3.	Containment Isolation					
a.	Phase A Isolation					
(1)	Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA
(2)	Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.13	NA
(3)	Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
b.	Phase B Isolation					
(1)	Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA
(2)	Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
(3)	Containment Pressure High - 3	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 28.3 psig

(a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.

Table 3.3.2-1 (page 3 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
4. Steam Line Isolation					
a. Manual Initiation	1,2 ⁽ⁱ⁾ , 3 ⁽ⁱ⁾	2	F	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2 ⁽ⁱ⁾ , 3 ⁽ⁱ⁾	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Automatic Actuation Logic and Actuation Relays (MSFIS)	1, 2 ⁽ⁱ⁾ , 3 ⁽ⁱ⁾	2 trains ^(o)	S	SR 3.3.2.3	NA
d. Containment Pressure - High 2	1,2 ⁽ⁱ⁾ , 3 ⁽ⁱ⁾	3	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 18.3 psig
e. Steam Line Pressure					
(1) Low	1,2 ⁽ⁱ⁾ , 3 ^{(b)(i)}	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 610 psig ^{(c)(s)}
(2) Negative Rate - High	3 ^{(g)(i)}	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 124 psi ^(h)

- (a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.
- (b) Above the P-11 (Pressurizer Pressure) Interlock and below P-11 unless the Function is blocked.
- (c) Time constants used in the lead/lag controller are $\tau_1 \geq 50$ seconds and $\tau_2 \leq 5$ seconds.
- (g) Below the P-11 (Pressurizer Pressure) Interlock; however, may be blocked below P-11 when safety injection on low steam line pressure is not blocked.
- (h) Time constant utilized in the rate/lag controller is ≥ 50 seconds.
- (i) Except when all MSIVs are closed.
- (o) Each train requires a minimum of two programmable logic controllers to be OPERABLE.
- (s) 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases.

Table 3.3.2-1 (page 4 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
5. Turbine Trip and Feedwater Isolation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2 ⁽ⁱ⁾ , 3 ⁽ⁱ⁾	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6 SR 3.3.2.14	NA
b. Automatic Actuation Logic and Actuation Relays (MSFIS)	1, 2 ⁽ⁱ⁾ , 3 ⁽ⁱ⁾	2 trains ^(o)	S	SR 3.3.2.3	NA
c. SG Water Level - High High (P-14)	1,2 ⁽ⁱ⁾	4 per SG	I	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 91.4% ^(s) of Narrow Range Instrument Span
d. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				

- (a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.
- (j) Except when all MFIVs are closed.
- (o) Each train requires a minimum of two programmable logic controllers to be OPERABLE.
- (s)
 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
 2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases.

Table 3.3.2-1 (page 5 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
5. Turbine Trip and Feedwater Isolation					
e. Steam Generator Water Level Low-Low ^(q)					
(1) Steam Generator Water Level Low-Low (Adverse Containment Environment)	1, 2 ^(j) , 3 ^(j)	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 20.6% ^(s) of Narrow Range Instrument Span
(2) Steam Generator Water Level Low-Low (Normal Containment Environment)	1 ^(r) , 2 ^(j,r) , 3 ^(j,r)	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 16.6% ^(s) of Narrow Range Instrument Span
(3) Not used.					
(4) Containment Pressure - Environmental Allowance Modifier	1, 2 ^(j) , 3 ^(j)	4	N	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 2.0 psig

- (a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.
- (j) Except when all MFIVs are closed.
- (k) Not used.
- (l) Not used.
- (q) Feedwater isolation only.
- (r) Except when the Containment Pressure – Environmental Allowance Modifier channels in the same protection sets are tripped.
- (s) 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases.

Table 3.3.2-1 (page 6 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
6. Auxiliary Feedwater					
a. Manual Initiation	1, 2, 3	1/pump	P	SR 3.3.2.8	NA
b. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.6	NA
c. Automatic Actuation Logic and Actuation Relays (BOP ESFAS)	1,2,3	2 trains	Q	SR 3.3.2.3	NA
d. SG Water Level Low-Low					
(1) Steam Generator Water Level Low-Low (Adverse Containment Environment)	1, 2, 3	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 20.6% ^(s) of Narrow Range Instrument Span
(2) Steam Generator Water Level Low-Low (Normal Containment Environment)	1 ^(r) , 2 ^(r) , 3 ^(r)	4 per SG	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 16.6% ^(s) of Narrow Range Instrument Span

- (a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.
- (r) Except when the Containment Pressure – Environmental Allowance Modifier channels in the same protection sets are tripped.
- (s) 1. If the as-found instrument channel setpoint is conservative with respect to the Allowable Value, but outside its as-found test acceptance criteria band, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service. If the as-found instrument channel setpoint is not conservative with respect to the Allowable Value, the channel shall be declared inoperable.
2. The instrument channel setpoint shall be reset to a value that is within the as-left setpoint tolerance band on either side of the Nominal Trip Setpoint, or to a value that is more conservative than the Nominal Trip Setpoint; otherwise, the channel shall be declared inoperable. The Nominal Trip Setpoints and the methodology used to determine the as-found test acceptance criteria band and the as-left setpoint tolerance band shall be specified in the Bases.

Table 3.3.2-1 (page 7 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
6. Auxiliary Feedwater					
d. SG Water Level Low-Low					
(3) Not used					
(4) Containment Pressure - Environmental Allowance Modifier	1, 2, 3	4	N	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ 2.0 psig
e. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
f. Loss of Offsite Power	1,2,3	2 trains	R	SR 3.3.2.7 SR 3.3.2.10	NA
g. Trip of all Main Feedwater Pumps	1,2 ⁽ⁿ⁾	2 per pump	J	SR 3.3.2.8	NA
h. Auxiliary Feedwater Pump Suction Transfer on Suction Pressure - Low	1,2,3	3	O	SR 3.3.2.1 SR 3.3.2.9 SR 3.3.2.10 SR 3.3.2.12	≥ 20.64 psia

(a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.

(k) Not used.

(l) Not used.

(n) Trip function may be blocked just before shutdown of the last operating main feedwater pump and restored just after the first main feedwater pump is put into service following performance of its startup trip test.

Table 3.3.2-1 (page 8 of 8)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE ^(a)
7. Automatic Switchover to Containment Sump					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.13	NA
b. Refueling Water Storage Tank (RWST) Level - Low Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ 35.2%
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.				
8. ESFAS Interlocks					
a. Reactor Trip, P-4	1,2,3	2 per train, 2 trains	F	SR 3.3.2.11	NA
b. Pressurizer Pressure, P-11	1,2,3	3	L	SR 3.3.2.5 SR 3.3.2.9	≤ 1981 psig
9. Automatic Pressurizer PORV Actuation					
a. Automatic Actuation Logic and Actuation Relays (SSPS)	1,2,3	2 trains	H	SR 3.3.2.2 SR 3.3.2.4 SR 3.3.2.14	NA
b. Pressurizer Pressure – High	1,2,3	4	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9	≤2350 psig

(a) The Allowable Value defines the limiting safety system setting except for Functions 1.e, 4.e.(1), 5.c, 5.e.(1), 5.e.(2), 6.d.(1), and 6.d.(2) (the Nominal Trip Setpoint defines the limiting safety system setting for these Functions). See the Bases for the Nominal Trip Setpoints.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

- a. Pressurizer pressure ≥ 2223 psig;
- b. RCS average temperature $\leq 590.1^{\circ}\text{F}$; and
- c. RCS total flow rate $\geq 382,630$ gpm.

APPLICABILITY: MODE 1.

-----NOTE-----
Pressurizer pressure limit does not apply during:

- a. THERMAL POWER ramp $> 5\%$ RTP per minute; or
 - b. THERMAL POWER step $> 10\%$ RTP.
-

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ 2223 psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is $\leq 590.1^{\circ}\text{F}$.	12 hours
SR 3.4.1.3	Verify RCS total flow rate is $\geq 382,630$ gpm.	12 hours
SR 3.4.1.4	<p>----- NOTE-----</p> <p>Calculated rather than verified by precision heat balance when performed prior to THERMAL POWER exceeding 75% RTP.</p> <p>-----</p> <p>Verify by precision heat balance that RCS total flow rate is $\geq 382,630$ gpm.</p>	<p>Once after each refueling prior to THERMAL POWER exceeding 75% RTP.</p> <p><u>AND</u></p> <p>18 months</p>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side narrow range water levels are $\geq 7\%$ for required RCS loops.	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required loops inoperable. <u>OR</u> No RCS or RHR loop in operation.	B.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u> B.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side narrow range water levels are $\geq 7\%$ for required RCS loops.	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

- LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:
- a. One additional RHR loop shall be OPERABLE; or
 - b. The secondary side wide range water level of at least two steam generators (SGs) shall be $\geq 86\%$.

-
- NOTES-----
1. The RHR pump of the loop in operation may be removed from operation for ≤ 1 hour per 8 hour period provided:
 - a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and
 - b. Core outlet temperature is maintained at least 10°F below saturation temperature.
 2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.
 3. No reactor coolant pump shall be started with any RCS cold leg temperature $\leq 275^\circ\text{F}$ unless the secondary side water temperature of each SG is $\leq 50^\circ\text{F}$ above each of the RCS cold leg temperatures.
 4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.
-

APPLICABILITY: MODE 5 with RCS loops filled.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify one RHR loop is in operation.	12 hours
SR 3.4.7.2	Verify SG secondary side wide range water level is $\geq 86\%$ in required SGs.	12 hours
SR 3.4.7.3	Verify correct breaker alignment and indicated power are available to the required RHR pump that is not in operation.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

- a. No pressure boundary LEAKAGE;
- b. 1 gpm unidentified LEAKAGE;
- c. 10 gpm identified LEAKAGE; and
- d. 150 gallons per day primary to secondary LEAKAGE through any one steam generator (SG).

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS operational LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE or primary to secondary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Pressure boundary LEAKAGE exists. <u>OR</u> Primary to secondary LEAKAGE not within limit.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	<p>-----NOTES-----</p> <p>1. Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>2. Not applicable to primary to secondary LEAKAGE.</p> <p>-----</p>	72 hours
	<p>Verify RCS operational LEAKAGE is within limits by performance of RCS water inventory balance.</p>	
SR 3.4.13.2	<p>-----NOTE-----</p> <p>Not required to be performed until 12 hours after establishment of steady state operation.</p> <p>-----</p>	72 hours
	<p>Verify primary to secondary LEAKAGE is ≤ 150 gallons per day through any one SG.</p>	

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.17 Steam Generator (SG) Tube Integrity

LCO 3.4.17 SG tube integrity shall be maintained.

AND

All SG tubes satisfying the tube repair criteria shall be plugged in accordance with Steam Generator Program.

APPLICABILITY: MODES 1 2, 3, and 4.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each SG tube.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more SG tubes satisfying the tube repair criteria and not plugged in accordance with the Steam Generator Program.	A.1 Verify tube integrity of the affected tube(s) is maintained until the next refueling outage or inspection.	7 days
	<u>AND</u> A.2 Plug the affected tube(s) in accordance with the Steam Generator Program.	Prior to entering MODE 4 following the next refueling outage or SG tube inspection

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> SG tube integrity not maintained.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.17.1	Verify SG tube integrity in accordance with the Steam Generator Program.	In accordance with the Steam Generator Program
SR 3.4.17.2	Verify that each inspected SG tube that satisfies the tube repair criteria is plugged in accordance with the Steam Generator Program.	Prior to entering MODE 4 following a SG tube inspection

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
4	≤ 85
3	≤ 45
2	≤ 27

5.5 Programs and Manuals (continued)

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 components. The program shall include the following:

- a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

<u>ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities</u>	<u>Required Frequencies for performing inservice testing activities</u>
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

- b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;
- c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and
- d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

5.5.9 Steam Generator (SG) Program

A Steam Generator Program shall be established and implemented to ensure that SG tube integrity is maintained. In addition, the Steam Generator Program shall include the following provisions:

- a. Provisions for condition monitoring assessments. Condition monitoring assessment means an evaluation of the "as found" condition of the tubing with respect to the performance criteria for structural integrity and accident induced leakage. The "as found" condition refers to the condition of the

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

tubing during a SG inspection outage, as determined from the inservice inspection results or by other means, prior to the plugging of tubes. Condition monitoring assessments shall be conducted during each outage during which the SG tubes are inspected or plugged to confirm that the performance criteria are being met.

- b. Performance criteria for SG tube integrity. SG tube integrity shall be maintained by meeting the performance criteria for tube structural integrity, accident induced leakage, and operational LEAKAGE.
 - 1. Structural integrity performance criterion: All inservice steam generator tubes shall retain structural integrity over the full range of normal operating conditions (including startup, operation in the power range, hot standby, and cooldown, and all anticipated transients included in the design specification) and design basis accidents. This includes retaining a safety factor of 3.0 (3 Δ P) against burst under normal steady state full power operation primary-to-secondary pressure differential and a safety factor of 1.4 against burst applied to the design basis accident primary-to-secondary pressure differentials. Apart from the above requirements, additional loading conditions associated with the design basis accidents, or combination of accidents in accordance with the design and licensing basis, shall also be evaluated to determine if the associated loads contribute significantly to burst or collapse. In the assessment of tube integrity, those loads that do significantly affect burst or collapse shall be determined and assessed in combination with the loads due to pressure with a safety factor of 1.2 on the combined primary loads and 1.0 on axial secondary loads.
 - 2. Accident induced leakage performance criterion: The primary to secondary accident induced leakage rate for any design basis accident, other than a SG tube rupture, shall not exceed the leakage rate assumed in the accident analysis in terms of total leakage rate for all SGs and leakage rate for an individual SG. Leakage is not to exceed 1 gpm total for all four steam generators.
 - 3. The operational LEAKAGE performance criterion is specified in LCO 3.4.13, "RCS Operational LEAKAGE."
- c. Provisions for SG tube repair criteria. Tubes found by inservice inspection to contain flaws with a depth equal to or exceeding 40% of the nominal tube wall thickness shall be plugged.

(continued)

5.5 Programs and Manuals (continued)

5.5.9 Steam Generator (SG) Program (continued)

- d. Provisions for SG tube inspections. Periodic SG tube inspections shall be performed. The number and portions of the tubes inspected and methods of inspection shall be performed with the objective of detecting flaws of any type (e.g., volumetric flaws, axial and circumferential cracks) that may be present along the length of the tube, from the tube-to-tubesheet weld at the tube inlet to the tube-to-tubesheet weld at the tube outlet, and that may satisfy the applicable tube repair criteria. The tube-to-tubesheet weld is not part of the tube. In addition to meeting the requirements of d.1, d.2, and d.3 below, the inspection scope, inspection methods, and inspection intervals shall be such as to ensure that SG tube integrity is maintained until the next SG inspection. An assessment of degradation shall be performed to determine the type and location of flaws to which the tubes may be susceptible and, based on this assessment, to determine which inspection methods need to be employed and at what locations.
1. Inspect 100% of the tubes in each SG during the first refueling outage following SG replacement.
 2. Inspect 100% of the tubes at sequential periods of 144, 108, 72, and, thereafter, 60 effective full power months. The first sequential period shall be considered to begin after the first inservice inspection of the SGs. In addition, inspect 50% of the tubes by the refueling outage nearest the midpoint of the period and the remaining 50% by the refueling outage nearest the end of the period. No SG shall operate for more than 72 effective full power months or three refueling outages (whichever is less) without being inspected.
 3. If crack indications are found in any SG tube, then the next inspection for each SG for the degradation mechanism that caused the crack indication shall not exceed 24 effective full power months or one refueling outage (whichever is less). If definitive information, such as from examination of a pulled tube, diagnostic non-destructive testing, or engineering evaluation indicates that a crack-like indication is not associated with a crack(s), then the indication need not be treated as a crack.
- e. Provisions for monitoring operational primary to secondary LEAKAGE.

(continued)

5.5 Programs and Manuals (continued)

5.5.10 Secondary Water Chemistry Program

This program provides controls for monitoring secondary water chemistry to inhibit SG tube degradation. The program shall include:

- a. Identification of a sampling schedule for the critical variables and control points for these variables;
- b. Identification of the procedures used to measure the values of the critical variables;
- c. Identification of process sampling points, which shall include monitoring the discharge of the condensate pumps for evidence of condenser in leakage;
- d. Procedures for the recording and management of data;
- e. Procedures defining corrective actions for all off control point chemistry conditions; and
- f. A procedure identifying the authority responsible for the interpretation of the data and the sequence and timing of administrative events, which is required to initiate corrective action.

5.5.11 Ventilation Filter Testing Program (VFTP)

A program shall be established to implement the following required testing of Engineered Safety Feature (ESF) filter ventilation systems at the frequencies specified in Regulatory Guide 1.52, Rev. 2, and uses the test procedure guidance in Regulatory Guide 1.52, Revision 2, Positions C.5.a, C.5.c and C.5.d.

- a. Demonstrate for each of the ESF systems that an inplace test of the high efficiency particulate air (HEPA) filters shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, \pm 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, \pm 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- b. Demonstrate for each of the ESF systems that an inplace test of the charcoal adsorber shows a penetration and system bypass < 1.0% when tested at the system flowrate specified below.

ESF Ventilation System	Flowrate
Control Room Filtration	2000 cfm, \pm 200 cfm
Control Room Pressurization	500 cfm, +500, -50 cfm
Emergency Exhaust System	9000 cfm, \pm 900 cfm

- c. Demonstrate for each of the ESF systems within 31 days after removal that a laboratory test of a sample of the charcoal adsorber, when obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, shows the methyl iodide penetration less than the value specified below when tested in accordance with ASTM D3803-1989 at a temperature of 30°C and the relative humidity specified below.

ESF Ventilation System	Penetration	RH
Control Room Filtration	2.0%	70%
Control Room Pressurization	2.0%	70%
Emergency Exhaust System	2.0%	70%

- d. Demonstrate at least once per 18 months for each of the ESF systems that the pressure drop across the combined HEPA filters and the charcoal adsorbers is less than the value specified below when tested at the system flowrate specified below.

ESF Ventilation System	Delta P	Flowrate
Control Room Filtration	5.4" WG	2000 cfm, \pm 200 cfm
Control Room Pressurization	5.4" WG	500 cfm, +500,- 50 cfm
Emergency Exhaust System	5.4" WG	9000 cfm, \pm 900 cfm

(continued)

5.5 Programs and Manuals

5.5.11 Ventilation Filter Testing Program (VFTP) (continued)

- e. Demonstrate at least once per 18 months that the heaters for each of the ESF systems dissipate the value specified below when tested in accordance with ANSI 510-1975 and corrected to design nameplate voltage settings.

ESF Ventilation System	Wattage
Control Room Pressurization	15 ± 2 KW
Emergency Exhaust System	37 ± 3 KW

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the VFTP test frequencies.

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program

This program provides controls for potentially explosive gas mixtures contained in the Gaseous Radwaste System, the quantity of radioactivity contained in gas storage tanks and the quantity of radioactivity contained in unprotected outdoor liquid storage tanks. The gaseous radioactivity quantities shall be determined following the methodology in Branch Technical Position (BTP) ETSB 11-5, "Postulated Radioactive Release due to Waste Gas System Leak or Failure, Revision 0". The liquid radwaste quantities shall be determined in accordance with Standard Review Plan, Section 15.7.3, "Postulated Radioactive Release due to Tank Failures, Revision 2".

The program shall include:

- a. The limits for concentrations of hydrogen and oxygen in the Gaseous Radwaste System and a surveillance program to ensure the limits are maintained. Such limits shall be appropriate to the system's design criteria (i.e., whether or not the system is designed to withstand a hydrogen explosion);
- b. A surveillance program to ensure that the quantity of radioactivity contained in each gas storage tank is less than the amount that would result in a whole body exposure of ≥ 0.5 rem to any individual in an unrestricted area, in the event of an uncontrolled release of the tanks' contents; and
- c. A surveillance program to ensure that the quantity of radioactivity contained in the outdoor liquid radwaste tanks listed below that are not

(continued)

5.5 Programs and Manuals

5.5.12 Explosive Gas and Storage Tank Radioactivity Monitoring Program (continued)

surrounded by liners, dikes, or walls, capable of holding the tanks' contents and that do not have tank overflows and surrounding area drains connected to the Liquid Radwaste System is less than the quantities determined in accordance with the Standard Review Plan, Section 15.7.3:

- a. Reactor Makeup Water Storage Tank,
- b. Refueling Water Storage Tank,
- c. Condensate Storage Tank, and
- d. Outside temporary tanks, excluding demineralizer vessels and the liner being used to solidify radioactive waste.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Explosive Gas and Storage Tank Radioactivity Monitoring Program surveillance frequencies.

5.5.13 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

- a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:
 - 1. an API gravity or an absolute specific gravity within limits,
 - 2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and
 - 3. a water and sediment content within limits for ASTM 2D fuel oil.
- b. Other properties for ASTM 2D fuel oil are analyzed within 31 days following sampling and addition of new fuel oil to storage tanks; and
- c. Total particulate concentration of the stored fuel oil is ≤ 10 mg/l when tested every 31 days based on applicable ASTM D-2276 standards.
- d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

(continued)

5.5 Programs and Manuals (continued)

5.5.14 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

- a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.
- b. Licensees may make changes to Bases without prior NRC approval provided the changes do not require either of the following:
 - 1. a change in the TS incorporated in the license; or
 - 2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59.
- c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.
- d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5.5.15 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6. The SFDP shall contain the following:

- a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;
- b. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

(continued)

5.5 Programs and Manuals

5.5.15 Safety Function Determination Program (SFDP) (continued)

- c. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and
- d. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

- a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or
- b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or
- c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

5.5.16 Containment Leakage Rate Testing Program

- a. 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, 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 exceptions:
 - 1. The visual examination of containment concrete surfaces intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWL, except where relief has been authorized by the NRC.

(continued)

5.5 Programs and Manuals

5.5.16 Containment Leakage Rate Testing Program (continued)

2. The visual examination of the steel liner plate inside containment intended to fulfill the requirements of 10 CFR 50, Appendix J, Option B testing, will be performed in accordance with the requirements of and frequency specified by ASME Section XI Code, Subsection IWE, except where relief has been authorized by the NRC.
 3. The unit is excepted from post-modification integrated leakage rate testing requirements associated with steam generator replacement during the Refuel 14 outage (fall of 2005).
- b. The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a , is 48.1 psig.
 - c. The maximum allowable containment leakage rate, L_a , at P_a , shall be 0.20% of the containment air weight per day.
 - d. Leakage rate acceptance criteria are:
 1. Containment leakage rate acceptance criterion is $\leq 1.0 L_a$. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $< 0.60 L_a$ for the Type B and C tests and $\leq 0.75 L_a$ for Type A tests;
 2. Air lock testing acceptance criteria are:
 - a) Overall air lock leakage rate is $\leq 0.05 L_a$ when tested at $\geq P_a$;
 - b) For each door, leakage rate is $\leq 0.005 L_a$ when pressurized to ≥ 10 psig.
 - e. The provisions of Technical Specification SR 3.0.2 do not apply to the test frequencies in the Containment Leakage Rate Testing Program.
 - f. The provisions of Technical Specification SR 3.0.3 are applicable to the Containment Leakage Rate Testing Program.
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5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Not used

5.6.2 Annual Radiological Environmental Operating Report

The Annual Radiological Environmental Operating Report covering the operation of the unit during the previous calendar year shall be submitted by May 1 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period.

The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

The Annual Radiological Environmental Operating Report shall include the results of analyses of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the locations specified in the table and figures in the ODCM, as well as summarized and tabulated results of these analyses and measurements in a format similar to the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the

(continued)

5.6 Reporting Requirements

5.6.2 Annual Radiological Environmental Operating Report (continued)

reasons for the missing results. The missing data shall be submitted in a supplementary report as soon as possible.

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit during the previous year shall be submitted prior to May 1 of each year, in accordance with 10 CFR 50.36a. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and Process Control Program and in conformance with 10 CFR 50.36a and 10 CFR Part 50, Appendix I, Section IV.B.1.

5.6.4 Not used

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

- a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:
 1. Moderator Temperature Coefficient limits in Specification 3.1.3,
 2. Shutdown Bank Insertion Limit for Specification 3.1.5,
 3. Control Bank Insertion Limits for Specification 3.1.6,
 4. Axial Flux Difference Limits for Specification 3.2.3,
 5. Heat Flux Hot Channel Factor, $F_Q(Z)$ F_Q^{RTP} , $K(Z)$, $W(Z)$ and F_Q Penalty Factors for Specification 3.2.1,
 6. Nuclear Enthalpy Rise Hot Channel Factor $F_{\Delta H}$, $F_{\Delta H}^{RTP}$, and Power Factor Multiplier, $PF_{\Delta H}$, limits for Specification 3.2.2.
 7. Shutdown Margin Limits for Specifications 3.1.1, 3.1.4, 3.1.5, 3.1.6, and 3.1.8.

(continued)

5.6 Reporting Requirements

5.6.5 CORE OPERATING LIMITS REPORT (COLR) (continued)

- b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
 - 1. WCAP-9272-P-A, "WESTINGHOUSE RELOAD SAFETY EVALUATION METHODOLOGY", July 1985 (W Proprietary).
 - 2. WCAP-10216-P-A, REV. 1A, "RELAXATION OF CONSTANT AXIAL OFFSET CONTROL AND FQ SURVEILLANCE TECHNICAL SPECIFICATION," February 1994 (W Proprietary).
 - 3. WCAP-10266-P-A, REV. 2, "THE 1981 VERSION OF WESTINGHOUSE EVALUATION MODEL USING BASH CODE," March 1987 (W Proprietary).
 - 4. NRC Safety Evaluation Reports dated July 1, 1991, "Acceptance for Referencing of Topical Report WCAP-12610 'VANTAGE + Fuel Assembly Reference Core Report' (TAC NO 77268)," and September 15, 1994, "Acceptance for Referencing of Topical Report WCAP-12610, Appendix B, Addendum 1, 'Extended Burnup Fuel Design Methodology and ZIRLO Fuel Performance Models' (TAC No. M86416)" (WCAP-12610-P-A).
- c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.
- d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

- a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing and PORV lift setting as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(continued)

5.6 Reporting Requirements

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (continued)

1. Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and
 2. Specification 3.4.12, "Cold Overpressure Mitigation System (COMS)."
- b. The analytical methods used to determine the RCS pressure and temperature and COMS PORV limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:
1. NRC letter, CALLAWAY PLANT, UNIT 1 – ISSUANCE OF AMENDMENT RE: PRESSURE TEMPERATURE LIMITS REPORT (TAC NOS. MA5631 and MA7287), dated March 24, 2000.
 2. WCAP-14040-NP-A, Revision 2, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves, January, 1996".
- c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

5.6.7 Not used.

5.6.8 PAM Report

When a report is required by Condition B or G of LCO 3.3.3, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.9 Not used.

(continued)

5.6 Reporting Requirements (continued)

5.6.10 Steam Generator Tube Inspection Report

A report shall be submitted within 180 days after the initial entry into MODE 4 following completion of an inspection performed in accordance with the Specification 5.5.9, Steam Generator (SG) Program. The report shall include:

- a. The scope of inspections performed on each SG;
 - b. Active degradation mechanisms found;
 - c. Nondestructive examination techniques utilized for each degradation mechanism;
 - d. Location, orientation (if linear), and measured sizes (if available) of service induced indications;
 - e. Number of tubes plugged during the inspection outage for each active degradation mechanism;
 - f. Total number and percentage of tubes plugged to date; and
 - g. The results of condition monitoring, including the results of tube pulls and in-situ testing.
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5.0 ADMINISTRATIVE CONTROLS

5.7 High Radiation Area

As provided in paragraph 20.1601(c) of 10 CFR Part 20, the following controls shall be applied to high radiation areas in place of the controls required by paragraph 20.1601 (a) and (b) of 10 CFR Part 20:

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation:

- a. Each entryway to such an area shall be barricaded and conspicuously posted as a high radiation area. Such barricades may be opened as necessary to permit entry or exit of personnel or equipment;
- b. Access to, and activities in, each such area shall be controlled by means of Radiation Work Permit (RWP) or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.
- c. Individuals qualified in radiation protection procedures and personnel continuously escorted by such individuals may be exempted from the requirement for an RWP or equivalent while performing their assigned duties provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
 1. A radiation monitoring device that continuously displays radiation dose rates in the area; or
 2. A radiation monitoring device that continuously integrates the radiation dose rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
 3. A radiation monitoring device that continuously transmits dose rate and cumulative dose rate information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area, or
 4. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and

(continued)

5.7 High Area Radiation Area

5.7.1 High Radiation Areas with Dose Rates Not Exceeding 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

- (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
 - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with individuals in the area who are covered by such surveillance.
- e. Except for individuals qualified in radiation protection procedures, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation:

- a. Each entryway to such an area shall be conspicuously posted as a high radiation area and shall be provided with a locked or continuously guarded door or gate that prevents unauthorized entry, and, in addition:
 - 1. All such door and gate keys shall be maintained under the administrative control of the Shift Supervisor/Operating Supervisor or Health Physics Supervision, or his or her designee.
 - 2. Doors and gates shall remain locked except during periods of personnel or equipment entry or exit.
- b. Access to, and activities in, each such area shall be controlled by means of an RWP or equivalent that includes specification of radiation dose rates in the immediate work area(s) and other appropriate radiation protection equipment and measures.

(continued)

5.7 High Area Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

- c. Individuals qualified in radiation protection procedures may be exempted from the requirement for an RWP or equivalent while performing radiation surveys in such areas provided that they are otherwise following plant radiation protection procedures for entry to, exit from, and work in such areas.
- d. Each individual or group entering such an area shall possess:
 - 1. A radiation monitoring device that continuously integrates the radiation rates in the area and alarms when the device's dose alarm setpoint is reached, with an appropriate alarm setpoint, or
 - 2. A radiation monitoring device that continuously transmits dose rate and cumulative dose information to a remote receiver monitored by radiation protection personnel responsible for controlling personnel radiation exposure within the area with the means to communicate with and control every individual in the area, or
 - 3. A self-reading dosimeter (e.g., pocket ionization chamber or electronic dosimeter) and
 - (i) Be under the surveillance, as specified in the RWP or equivalent, while in the area, of an individual qualified in radiation protection procedures, equipped with a radiation monitoring device that continuously displays radiation dose rates in the area; who is responsible for controlling personnel exposure within the area, or
 - (ii) Be under the surveillance as specified in the RWP or equivalent, while in the area, by means of closed circuit television, of personnel qualified in radiation protection procedures, responsible for controlling personnel radiation exposure in the area, and with the means to communicate with and control every individual in the area, or
 - 4. In those cases where options (2) and (3), above, are impractical or determined to be inconsistent with the "As Low As is Reasonably

(continued)

5.7 High Area Radiation Area

5.7.2 High Radiation Areas with Dose Rates Greater than 1.0 rem/hour at 30 Centimeters from the Radiation Source or from any Surface Penetrated by the Radiation, but less than 500 rads/hour at 1 Meter from the Radiation Source or from any Surface Penetrated by the Radiation: (continued)

Achievable" principle, a radiation monitoring device that continuously displays radiation dose rates in the area.

- e. Except for individual qualified in radiation protection procedures or personnel continuously escorted by such individuals, entry into such areas shall be made only after dose rates in the area have been determined and entry personnel are knowledgeable of them.
 - f. Such individual areas that are within a larger area, such as PWR containment, where no enclosure exists for the purpose of locking and where no enclosure can reasonably be constructed around the individual area need not be controlled by a locked door or gate nor continuously guarded, but shall be barricaded, conspicuously posted, and a clearly visible flashing light shall be activated at the area as a warning device.
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