ENCLOSURE 1

TECHNICAL SPECIFICATIONS - REPLACEMENT PAGES

TECHNICAL SPECIFICATIONS BASES - REPLACEMENT PAGES

WATTS BAR NUCLEAR PLANT
TECHNICAL SPECIFICATIONS

INSTRUCTION SHEET FOR PAGE REPLACEMENT

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LIST OF ACRONYMS

Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ARV	Atmospheric Relief Valve
BOC	Beginning of Cycle
CAOC	Constant Axial Offset Control
CCS	Component Cooling Water System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
EOC	End of Cycle
ERCW	Essential Raw Cooling Water
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilating, and Air-Conditioning
LCO	Limiting Condition For Operation
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MSIV	Main Steam Line Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NMS	Neutron Monitoring System
ODCM	Offsite Dose Calculation Manual
PCP	Process Control Program
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PTLR	Pressure and Temperature Limits Report
QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power
RTS	Reactor Trip System
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.3	-------------------NOTES------------------- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Required to be performed within 96 hours after THERMAL POWER is ≥ 15% RTP. --- Compare results of the incore detector measurements to NIS AFD.	31 effective full power days (EFPD)
SR 3.3.1.4	-------------------NOTE------------------- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. --- Perform TADOT.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	Perform ACTUATION LOGIC TEST.	31 days on a STAGGERED TEST BASIS
SR 3.3.1.6	-------------------NOTE------------------- Required to be performed within 6 days after THERMAL POWER is ≥ 50% RTP. --- Calibrate excore channels to agree with incore detector measurements.	92 EFPD

(continued)

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.1.7	------------------NOTE------------------ For Functions 2 and 3 (Power Range Instrumentation), this Surveillance shall include verification that interlock P-10 is in the required state for existing unit conditions. -- Perform COT.	 92 days
SR 3.3.1.8	------------------NOTES------------------ 1. Not required to be performed for Source Range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. 2. This Surveillance shall include verification that interlock P-6 is in the required state for existing unit conditions. -- Perform COT.	------NOTE----- Only required when not performed within previous 31 days --------------- Prior to reactor startup AND Four hours after reducing power below P-10 for intermediate range instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND Every 31 days thereafter

(continued)

Table 3.3.1-1 (page 1 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA	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 3.3.1.15	≤ 111.4% RTP	109% RTP
b. Low	1(b),2	4	E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 SR 3.3.1.15	≤ 27.4% RTP	25% RTP
3. Power Range Neutron Flux Rate						
a. High Positive Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
b. High Negative Rate	1,2	4	E	SR 3.3.1.7 SR 3.3.1.11	≤ 6.3% RTP with time constant ≥ 2 sec	5% RTP with time constant ≥ 2 sec
4. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 40% RTP	25% RTP
	2(d)	2	H	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 40% RTP	25% RTP

(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

Table 3.3.1-1 (page 2 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
5. Source Range Neutron Flux	2(d)	2	I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.5 E5 cps	1.0 E5 cps
	3(a), 4(a), 5(a)	2	J,K	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 1.5 E5 cps	1.0 E5 cps
	3(e), 4(e), 5(e)	1	L	SR 3.3.1.1 SR 3.3.1.11	N/A	N/A
6. Overtemperature ΔT	1,2	4	W	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 3.3.1.15	Refer to Note 1 (Page 3.3-21)	Refer to Note 1 (Page 3.3-21)
7. Overpower ΔT	1,2	4	W	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	Refer to Note 2 (Page 3.3-22)	Refer to Note 2 (Page 3.3-22)
8. Pressurizer Pressure						
a. Low	1(f)	4	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 1964.8 psig	1970 psig
b. High	1,2	4	W	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≤ 2390.2 psig	2385 psig

(continued)

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication.

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

Table 3.3.1-1 (page 3 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
9. Pressurizer Water Level – High	1(f)	3	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ 92.7% span	92% span
10. Reactor Coolant Flow – Low						
a. Single Loop	1(g)	3 per loop	N	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.6% flow	90% flow
b. Two Loops	1(h)	3 per loop	X	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 89.6% flow	90% flow
11. Undervoltage RCPs	1(f)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 4734 V	4830 V
12. Underfrequency RCPs	1(f)	1 per bus	M	SR 3.3.1.9 SR 3.3.1.10 SR 3.3.1.15	≥ 56.9 Hz	57.5 Hz

(continued)

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

(g) Above the P-8 (Power Range Neutron Flux) interlock.

(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

Table 3.3.1-1 (page 4 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
13. SG Water Level--Low-low	1,2	3/SG	U	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 SR 3.3.1.15	≥ 16.4% of narrow range span	17% of narrow range span
Coincident with:						
a) Vessel ΔT Equivalent to power ≤ 50% RTP	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input 50% RTP
With a time delay (Ts) if one steam generator is affected					≤ 1.01 Ts (Refer to Note 3, Page 3.3-23)	Ts (Refer to Note 3, Page 3.3-23)
or						
A time delay (Tm) if two or more Steam Generators are affected					≤ 1.01 Tm (Refer to Note 3, Page 3.3-23)	Tm (Refer to Note 3, Page 3.3-23)
OR						
b) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	V	SR 3.3.1.7 SR 3.3.1.10	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input 50% RTP
14. Turbine Trip						
a. Low Fluid Oil pressure	1(i)	3	O	SR 3.3.1.10 SR 3.3.1.14	≥ 43 psig	45 psig
b. Turbine Stop Valve Closure	1(i)	4	Y	SR 3.3.1.10 SR 3.3.1.14	≥ 1% open	1% open

(continued)

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

Table 3.3.1-1 (page 5 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
15. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	P	SR 3.3.1.13	NA	NA
16. Reactor Trip System Interlocks						
a. Intermediate Range Neutron Flux, P-6						
(1) Enable Manual Block of SR Trip	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	NA	1.66E-04% RTP
(2) Auto Reset (Unblock Manual Block of SR Trip)	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	≥ 7.65E-5% RTP	0.47E-4% RTP below setpoint
b. Low Power Reactor Trips Block, P-7	1	1 per train	S	SR 3.3.1.11 SR 3.3.1.12	NA	NA
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 50.4% RTP	48% RTP
d. Power Range Neutron Flux, P-9	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 52.4% RTP	50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	R	SR 3.3.1.11 SR 3.3.1.12	≥ 7.6% RTP and ≤ 12.4% RTP	10% RTP
f. Turbine Impulse Pressure, P-13	1	2	S	SR 3.3.1.10 SR 3.3.1.12	≤ 12.4% full-power pressure	10% full-power pressure

(continued)

(d) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

Table 3.3.1-1 (page 6 of 9)
Reactor Trip System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
17. Reactor Trip Breakers[j]	1,2	2 trains	Q	SR 3.3.1.4	NA	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.4	NA	NA
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	T	SR 3.3.1.4	NA	NA
	3[a], 4[a], 5[a]	1 each per RTB	C	SR 3.3.1.4	NA	NA
19. Automatic Trip Logic	1,2	2 trains	P	SR 3.3.1.5	NA	NA
	3[a], 4[a], 5[a]	2 trains	C	SR 3.3.1.5	NA	NA

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(j) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.2.10	-------------------NOTE------------------- Required to be performed for the turbine driven AFW pump within 24 hours after SG pressure is ≥ 1092 psig. -- Verify ESFAS RESPONSE TIMES are within limit.	18 months on a STAGGERED TEST BASIS
SR 3.3.2.11	------------------NOTE-------------------- Verification of setpoint not required. -- Perform TADOT.	Once per reactor trip breaker cycle

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

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5 SR 3.3.2.7	NA	NA
c. Containment Pressure – High	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 1.6 psig	1.5 psig
d. Pressurizer Pressure – Low	1,2,3[(a)]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 1864.8 psig	1870 psig
e. Steam Line Pressure-Low	1,2,3[(a)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 666.6[(b)] psig	675[(b)] psig
2. Containment Spray						
a. Manual Initiation	1,2,3,4	2 per train, 2 trains	B	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure--High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 2.9 psig	2.8 psig

(continued)

(a) Above the P-11 (Pressurizer Pressure) Interlock.

(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.

Table 3.3.2-1 (page 2 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
3. Containment Isolation						
a. Phase A Isolation						
(1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5 SR 3.3.2.7	NA	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	NA
(2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5 SR 3.3.2.7	NA	NA
(3) Containment Pressure--High High	1,2,3	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 2.9 psig	2.8 psig
4. Steam Line Isolation						
a. Manual Initiation	1,2[(c)],3[(c)]	1/valve	F	SR 3.3.2.8	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2[(c)],3[(c)]	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA

(continued)

(c) Except when all MSIVs are closed and de-activated.

Table 3.3.2-1 (page 3 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
4. Steam Line Isolation (continued)						
c. Containment Pressure-High High	1,2(c),3(c)	4	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 2.9 psig	2.8 psig
d. Steam Line Pressure						
(1) Low	1,2(c), 3(a)(c)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 666.6(b) psig	675(b) psig
(2) Negative Rate-High	3(d)(c)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 108.5(e) psi	100(e) psi
5. Turbine Trip and Feedwater Isolation						
a. Automatic Actuation Logic and Actuation Relays	1,2(f), 3(f)	2 trains	H	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level – High High (P-14)	1,2(f), 3(f)	3 per SG	I	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≤ 83.1%	82.4%
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
d. North MSV Vault Room Water Level - High	1,2(f)(g)	3/vault room	O	SR 3.3.2.6 SR 3.3.2.9	≤ 5.31 inches	4 inches
e. South MSV Vault Room Water Level - High	1,2(f)(g)	3/vault room	O	SR 3.3.2.6 SR 3.3.2.9	≤ 4.56 inches	4 inches

(continued)

(a) Above the P-11 (Pressurizer Pressure) interlock.
(b) Time constants used in the lead/lag controller are $t_1 \geq 50$ seconds and $t_2 \leq 5$ seconds.
(c) Except when all MSIVs are closed and de-activated.
(d) Function automatically blocked above P-11 (Pressurizer Interlock) setpoint and is enabled below P-11 when safety injection on Steam Line Pressure Low is manually blocked.
(e) Time constants utilized in the rate/lag controller are t_3 and $t_4 \geq 50$ seconds.
(f) Except when all MFIVs, MFRVs, and associated bypass valves are closed and de-activated or isolated by a closed manual valve.
(g) MODE 2 if Turbine Driven Main Feed Pumps are operating.

Table 3.3.2-1 (page 4 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater						
a. Automatic Actuation Logic and Actuation Relays	1,2,3	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
b. SG Water Level-Low Low	1,2,3	3 per SG	M	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 16.4%	17.0%
Coincident with:						
1) Vessel ΔT equivalent to power ≤ 50% RTP	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input 50% RTP
With a time delay (Ts) if one S/G is affected					≤ 1.01 Ts (Note 1, Page 3.3-40)	Ts (Note 1, Page 3.3-40)
or						
A time delay (Tm) if two or more S/G's are affected					≤ 1.01 Tm (Note 1, Page 3.3-40)	Tm (Note 1, (Page 3.3-40)
<u>OR</u>						
2) Vessel ΔT equivalent to power > 50% RTP with no time delay (Ts and Tm = 0)	1,2	3	N	SR 3.3.2.4 SR 3.3.2.9	Vessel ΔT variable input ≤ 52.7% RTP	Vessel ΔT variable input 50% RTP

(continued)

Table 3.3.2-1 (page 5 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
6. Auxiliary Feedwater (continued)						
c. Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements					
d. Loss of Offsite Power	1,2,3	4 per bus	F	Refer to Function 4 of Table 3.3.5-1 for SRs and Allowable Values		
e. Trip of all Main Feedwater Pumps	1,2	1 per pump	J	SR 3.3.2.8 SR 3.3.2.9 SR 3.3.2.10	≥ 48 psig	50 psig
f. Motor-Driven Auxiliary Feedwater Pumps Train A and B Suction Transfer on Suction Pressure —Low	1,2,3	3	F	SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	A) ≥ 0.5 psig B) ≥ 1.33 psig	A) 1.2 psig B) 2.0 psig
g. Turbine-driven AFW Pump Suction Train A and B Transfer on Suction Pressure--Low	1,2,3	3/train, 2 trains	F	SR 3.3.2.6 SR 3.3.2.9 SR 3.3.2.10	A) ≥ 11.6 psig B) ≥ 12.2 psig	A) 12.8 psig B) 13.5 psig
7. Automatic Switchover to Containment Sump						
a. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA

(continued)

Table 3.3.2-1 (page 6 of 7)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	NOMINAL TRIP SETPOINT
7. Automatic Switchover to Containment Sump (continued)						
b. Refueling Water Storage Tank (RWST) Level – Low	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 155.6 inches from Tank Base	158 inches from Tank Base
Coincident with Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
and						
Coincident with Containment Sump Level – High	1,2,3,4	4	K	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9 SR 3.3.2.10	≥ 37.2 in. above el. 702.8 ft	38.2 in. above el. 702.8 ft
8. ESFAS Interlocks						
a. Reactor Trip, P-4	1,2,3	1 per train, 2 trains	F	SR 3.3.2.11	NA	NA
b. Pressurizer Pressure, P-11						
(1) Unblock (Auto Reset of SI Block)	1,2,3	3	L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9	≤ 1975.2 psig	1970 psig
(2) Enable Manual Block of SI	1,2,3	3	L	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.9	≥1956.8 psig	1962 psig

Table 3.3.2-1 (page 7 of 7)

Engineered Safety Feature Actuation System Instrumentation

NOTE 1: Steam Generator Water Level Low-Low Trip Time Delay:

$$T_s = A(P)^3 + B(P)^2 + C(P) + D$$

$$T_m = E(P)^3 + F(P)^2 + G(P) + H$$

Where:

P = Vessel ΔT Equivalent to power (% RTP), P $\leq$ 50% RTP.

T_s = Time Delay for Steam Generator Water Level--Low-Low Reactor Trip, one Steam Generator affected.

T_m = Time Delay for Steam Generator Water Level--Low-Low Reactor Trip, two or more Steam Generators affected.

A = -0.0085041	E = -0.0047421
B = 0.9266400	F = 0.5682600
C = -33.85998	G = -23.70753
D = 474.6060	H = 357.9840

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ 40 gpm with charging pump discharge header pressure ≥ 2430 psig and the pressurizer level control valve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Seal injection flow not within limit.	A.1	Adjust manual seal injection throttle valves to give a flow within limit with charging pump discharge header pressure ≥ 2430 psig and the pressurizer level control valve full open.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.5.1	-------------------NOTE------------------- Required to be performed within 4 hours after the Reactor Coolant System pressure stabilizes at ≥ 2215 psig and ≤ 2255 psig. -- Verify manual seal injection throttle valves are adjusted to give a flow within limit with charging pump discharge header pressure ≥ 2430 psig and the pressurizer level control valve full open.	31 days

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.6.2.1	-------------------NOTES------------------- 1. An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. 2. Results shall be evaluated against acceptance criteria of SR 3.6.1.1 in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. --- Perform required air lock leakage rate testing in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions. The acceptance criteria for air lock testing are: a. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at ≥ 15.0 psig. b. For each door, leakage rate is $\leq 0.01\ L_a$ when tested at ≥ 6 psig.	-----NOTE------ SR 3.0.2 is not applicable -------------- In accordance with 10 CFR 50, Appendix J, as modified by approved exemptions
SR 3.6.2.2	-------------------NOTE------------------- Only required to be performed upon entry or exit through the containment air lock. --- Verify only one door in the air lock can be opened at a time.	184 days

3.6 CONTAINMENT SYSTEMS

3.6.3 Containment Isolation Valves

LCO 3.6.3 Each containment isolation valve shall be OPERABLE.

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

ACTIONS

-----------------------------------NOTES-----------------------------------

1. Penetration flow path(s) may be unisolated intermittently under administrative controls.
2. Separate Condition entry is allowed for each penetration flow path.
3. Enter applicable Conditions and Required Actions for systems made inoperable by containment isolation valves.
4. Enter applicable Conditions and Required Actions of LCO 3.6.1, "Containment," when isolation valve leakage results in exceeding the overall containment leakage rate acceptance criteria.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Only applicable to penetration flow paths with two containment isolation valves. -------------------- One or more penetration flow paths with one containment isolation valve inoperable except for purge valve or shield building bypass leakage not within limit.	A.1 Isolate the affected penetration flow path by use of at least one closed and de-activated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. AND	4 hours

(continued)

3.7 PLANT SYSTEMS

3.7.11 Control Room Emergency Air Temperature Control System (CREATCS)

LCO 3.7.11 Two CREATCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, 4, 5, and 6
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CREATCS train inoperable.	A.1 Restore CREATCS train to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met in MODE 1, 2, 3, or 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours
C. Required Action and associated Completion Time of Condition A not met in MODE 5 or 6, or during movement of irradiated fuel assemblies.	C.1 Place OPERABLE CREATCS train in operation. OR C.2.1 Suspend CORE ALTERATIONS. AND C.2.2 Suspend movement of irradiated fuel assemblies.	Immediately Immediately Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two CREATCS trains inoperable in MODE 5 or 6, or during movement of irradiated fuel assemblies.	D.1	Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>		
	D.2	Suspend movement of irradiated fuel assemblies.	Immediately
E. Two CREATCS trains inoperable in MODE 1, 2, 3, or 4.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.11.1	Verify each CREATCS train has the capability to remove the assumed heat load.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.12	-------------------NOTE------------------- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. --	
	Verify on an actual or simulated Engineered Safety Feature (ESF) actuation signal each Unit 1 DG auto-starts from standby condition and: a. In ≤ 10 seconds after auto-start and during tests, achieves voltage ≥ 6800 V and frequency ≥ 58.8 Hz; b. After DG fast start from standby conditions the DG achieves steady state voltage ≥ 6800 V and ≤ 7260 V, and frequency ≥ 58.8 Hz and ≤ 61.2 Hz. c. Operates for ≥ 5 minutes; d. Permanently connected loads remain energized from the offsite power system; and e. Emergency loads are energized from the offsite power system.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.13	------------------NOTE------------------ This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. -- Verify each DG's automatic trips are bypassed on automatic or emergency start signal except: a. Engine overspeed; and b. Generator differential current.	18 months
SR 3.8.1.14	------------------NOTES------------------ 1. Momentary transients outside the load and power factor ranges do not invalidate this test. 2. This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. -- Verify each DG operating at a power factor $\geq$ 0.8 and $\leq$ 0.9 operates for $\geq$ 24 hours: a. For $\geq$ 2 hours loaded $\geq$ 4620 kW and $\leq$ 4840 kW and $\geq$ 3465 kVAR and $\leq$ 3630 kVAR; and b. For the remaining hours of the test loaded $\geq$ 3960 kW and $\leq$ 4400 kW and $\geq$ 2970 kVAR and $\leq$ 3300 kVAR.	18 months

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.7 Inverters – Operating

LCO 3.8.7 Two inverters in each of four channels shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more inverters in one channel inoperable.	A.1	---------NOTE--------- Enter applicable Conditions and Required Actions of LCO 3.8.9, "Distribution Systems-Operating", with any AC Vital Bus deenergized. --------------------- Restore inverter(s) to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.7.1	Verify correct inverter voltage, frequency, and alignment to required AC vital bus and from associated vital battery board and 480 V shutdown board.	7 days

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

g) Preservice Inspection - An inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections.

h) Tube Inspection - An inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

i) Unserviceable - The condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operational Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break accident as specified in Specification 5.7.2.12.e.

2. The SG shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.7.2.12-1.

g. <u>Reports</u> - The content and frequency of written reports shall be in accordance with Specification 5.9.9.

(continued)

5.7 Procedures, Programs, and Manuals (continued)

TABLE 5.7.2.12-1

STEAM GENERATOR TUBE INSPECTION
SUPPLEMENTAL SAMPLING REQUIREMENTS

1st Sample Inspection			2nd Sample Inspection		3rd Sample Inspection	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S tubes per SG	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug defective tubes and inspect an additional 2S tubes in this SG.	C-1	None	N/A	N/A
			C-2	Plug defective tubes and inspect an additional 4S tubes in this SG.	C-1	N/A
					C-2	Plug defective tubes.
					C-3	Perform action for C-3 result of first sample.
			C-3	Perform action for C-3 result of first sample.	N/A	N/A
	C-3	Inspect all tubes in this SG, plug defective tubes and inspect 2S tubes in each other SG. Notification to NRC pursuant to 10CFR50.72	All other SGs C-1	None	N/A	N/A
			Some SGs C-2 but no other is C-3	Perform action for C-2 result of second sample.	N/A	N/A
			Additional SG is C-3	Inspect all tubes in each SG and plug defective tubes. Notification to NRC pursuant to 10CFR50.72.	N/A	N/A

S = 3 N/n % Where N is the number of SGs in the unit and n is the number of S.G.s inspected during an inspection.

(continued)

WATTS BAR NUCLEAR PLANT
TECHNICAL SPECIFICATIONS - BASES

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LIST OF FIGURES

LIST OF ACRONYMS

Acronym	Title
ABGTS	Auxiliary Building Gas Treatment System
ACRP	Auxiliary Control Room Panel
ASME	American Society of Mechanical Engineers
AFD	Axial Flux Difference
AFW	Auxiliary Feedwater System
ARO	All Rods Out
ARFS	Air Return Fan System
ADV	Atmospheric Dump Valve
BOC	Beginning of Cycle
CAOC	Constant Axial Offset Control
CCS	Component Cooling System
CFR	Code of Federal Regulations
COLR	Core Operating Limits Report
CREVS	Control Room Emergency Ventilation System
CSS	Containment Spray System
CST	Condensate Storage Tank
DNB	Departure from Nucleate Boiling
ECCS	Emergency Core Cooling System
EFPD	Effective Full-Power Days
EGTS	Emergency Gas Treatment System
EOC	End of Cycle
ERCW	Essential Raw Cooling Water
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
HEPA	High Efficiency Particulate Air
HVAC	Heating, Ventilating, and Air-Conditioning
LCO	Limiting Condition For Operation
MFIV	Main Feedwater Isolation Valve
MFRV	Main Feedwater Regulation Valve
MSIV	Main Steam Line Isolation Valve
MSSV	Main Steam Safety Valve
MTC	Moderator Temperature Coefficient
NMS	Neutron Monitoring System
ODCM	Offsite Dose Calculation Manual
PCP	Process Control Program
PIV	Pressure Isolation Valve
PORV	Power-Operated Relief Valve
PTLR	Pressure and Temperature Limits Report
QPTR	Quadrant Power Tilt Ratio
RAOC	Relaxed Axial Offset Control
RCCA	Rod Cluster Control Assembly
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RTP	Rated Thermal Power
RTS	Reactor Trip System

LIST OF ACRONYMS
(Page 2 of 2)

Acronym	Title
RWST	Refueling Water Storage Tank
SG	Steam Generator
SI	Safety Injection
SL	Safety Limit
SR	Surveillance Requirement
UHS	Ultimate Heat Sink

TECHNICAL SPECIFICATIONS BASES

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Automatic Switchover to Containment Sump (continued)

Switchover from the RWST to the containment sump must occur before the RWST empties to prevent damage to the RHR pumps and a loss of core cooling capability. For similar reasons, switchover must not occur before there is sufficient water in the containment sump to support ESF pump suction. Furthermore, early switchover must not occur to ensure that sufficient borated water is injected from the RWST. This ensures the reactor remains shut down in the recirculation mode.

a. Automatic Switchover to Containment Sump-Automatic Actuation Logic and Actuation Relays

Automatic actuation logic and actuation relays consist of the same features and operate in the same manner as described for ESFAS Function 1.b.

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Coincident With Safety Injection and Coincident With Containment Sump Level-High

During the injection phase of a LOCA, the RWST is the source of water for all ECCS pumps. A low level in the RWST coincident with an SI signal provides protection against a loss of water for the ECCS pumps and indicates the end of the injection phase of the LOCA. The RWST is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability.

The RWST – Low Trip Setpoint is selected to ensure switchover occurs before the RWST empties, to prevent ECCS pump damage.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Coincident With Safety Injection and Coincident With Containment Sump Level-High (continued)

This setpoint will also ensure that enough borated water is injected to maintain the reactor shut down. The limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction.

The transmitters are located in an area not affected by HELBs or post accident high radiation. Thus, they will not experience any adverse environmental conditions and the Trip Setpoint reflects only steady state instrument uncertainties.

Automatic switchover occurs only if the RWST low level signal is coincident with SI. This prevents accidental switchover during normal operation. Accidental switchover could damage ECCS pumps if they are attempting to take suction from an empty sump. The automatic switchover Function requirements for the SI Functions are the same as the requirements for their SI function. Therefore, the requirements are not repeated in Table 3.3.2-1. Instead, Function 1, SI, is referenced for all initiating Functions and requirements.

Additional protection from spurious switchover is provided by requiring a Containment Sump Level – High signal as well as RWST Level – Low and SI. This ensures sufficient water is available in containment to support the recirculation phase of the accident. A Containment Sump Level – High signal must be present, in addition to the SI

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

b. Automatic Switchover to Containment Sump-Refueling Water Storage Tank (RWST) Level-Low Coincident With Safety Injection and Coincident With Containment Sump Level-High (continued)

signal and the RWST Level – Low signal, to transfer the suctions of the RHR pumps to the containment sump. The containment sump is equipped with four level transmitters. These transmitters provide no control functions. Therefore, a two-out-of-four logic is adequate to initiate the protection function actuation. Although only three channels would be sufficient, a fourth channel has been added for increased reliability. The containment sump level Trip Setpoint is selected to ensure enough borated water is injected to ensure the reactor remains shut down. The high limit also ensures adequate water inventory in the containment sump to provide ECCS pump suction. The transmitters are located inside containment and thus possibly experience adverse environmental conditions. Therefore, the Trip Setpoint reflects the inclusion of both steady state and environmental instrument uncertainties.

These Functions must be OPERABLE in MODES 1, 2, 3, and 4 when there is a potential for a LOCA to occur, to ensure a continued supply of water for the ECCS pumps. These Functions are not required to be OPERABLE in MODES 5 and 6 because there is adequate time for the operator to evaluate unit conditions and respond by manually starting systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. System pressure and temperature are very low and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

8. Engineered Safety Feature Actuation System Interlocks

To allow some flexibility in unit operations, several interlocks are included as part of the ESFAS. These interlocks permit the operator to block some signals, automatically enable other signals, prevent some actions from occurring, and cause other actions to occur. The interlock Functions back up manual actions to ensure bypassable functions are in operation under the conditions assumed in the safety analyses.

a. Engineered Safety Feature Actuation System Interlocks-Reactor Trip, P-4

The P-4 interlock is enabled when a reactor trip breaker (RTB) and its associated bypass breaker is open. Once the P-4 interlock is enabled, automatic SI initiation may be blocked after a 90 second time delay. This Function allows operators to take manual control of SI systems after the initial phase of injection is complete. Once SI is blocked, automatic actuation of SI cannot occur until the RTBs have been manually closed. The functions of the P-4 interlock are:

- Trip the main turbine;
- Isolate MFW with coincident low T_{avg};
- Prevent reactuation of SI after a manual reset of SI;
- Transfer the steam dump from the load rejection controller to the unit trip controller; and
- Prevent opening of the MFW isolation valves if they were closed on SI or SG Water Level – High High, or MSVV Water Level - High.

Each of the above Functions is interlocked with P-4 to avert or reduce the continued cooldown of the RCS following a reactor trip. An excessive cooldown of the RCS following a reactor trip

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

In the accident analysis presented in Chapter 15.0 of the FSAR (Ref. 2), the ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is also required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event and also for other accidents. Thus, the SGTR is the limiting event for the ADVs. Four ADVs are required to be OPERABLE to satisfy the SGTR accident analysis requirements. This considers any single failure assumptions regarding the failure of one ADV to open on demand.

The ADVs are equipped with block valves in the event an ADV spuriously fails to open or fails to close during use.

The ADVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

Four ADV lines are required to be OPERABLE. One ADV line is required from each of four steam generators to ensure that at least two ADV lines are available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable, accompanied by a single, active failure of a second ADV line on an unaffected steam generator. The block valves must be OPERABLE to isolate a failed open ADV line.

Failure to meet the LCO can result in a delay in completing the SGTR recovery operations which could result in dose consequences that exceed accident analysis criteria.

(continued)

BASES

LCO (continued)	An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.
APPLICABILITY	In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE. In MODE 5 or 6, an SGTR is not a credible event.
ACTIONS	A.1

ACTIONS

A.1

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines, a nonsafety grade backup in the Steam Dump System, and MSSVs. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

B.1

With two or more ADV lines inoperable, action must be taken to restore all but one ADV line to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Dump System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

C.1 and C.2

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the plant must be placed in a MODE in which the LCO does not apply. To achieve this status, the plant must be placed in at least

(continued)

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT
TECHNICAL REQUIREMENTS MANUAL

INSTRUCTION SHEET FOR PAGE REPLACEMENT

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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	C.1 Restore RWST to OPERABLE status.	8 hours
D. RWST inoperable for reasons other than Condition C.	D. 1 Restore RWST to OPERABLE status.	1 hour
E. Required Action and associated Completion Time of Condition C or D not met.	E.1 Be in MODE 3 AND E.2 Be in MODE 4 with one or more RCS cold leg temperatures ≤ 310 °F.	6 hours 12 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
TSR 3.1.6.1	-----------------NOTE----------------- Only required when outside air temperature is < 60 °F or >105 °F. -------------------------------------- Verify RWST solution temperature is ≥ 60 °F and ≤ 105 °F.	24 hours
TSR 3.1.6.2	Verify RWST boron concentration is ≥ 2,000 ppm and ≤ 2,100 ppm.	7 days

(continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.1.6.3	Verify RWST borated water volume is ≥ 370,000 gallons.	7 days
TSR 3.1.6.4	-----------------NOTE----------------- Only required if the BAT is required OPERABLE. -------------------------------------- Verify Boric Acid Tank (BAT) solution temperature is ≥ 63°F.	24 hours
TSR 3.1.6.5	-----------------NOTE----------------- Only required if the BAT is required OPERABLE. -------------------------------------- Verify BAT boron concentration is in accordance with Figure 3.1.6.	7 days
TSR 3.1.6.6	-----------------NOTE----------------- Only required if the BAT is required OPERABLE. -------------------------------------- Verify BAT borated water volume is in accordance with Figure 3.1.6.	7 days

Table 3.3.2-1 (Page 1 of 5)

Engineered Safety Features Actuation System Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
1. Manual Initiation	
a. Safety Injection (ECCS)	N.A.
b. Containment Spray	N.A.
c. Phase "A" Isolation	N.A.
d. Phase "B" Isolation	N.A.
e. Containment Ventilation Isolation	N.A.
f. Steam Line Isolation	N.A.
g. Feedwater Isolation	N.A.
h. Auxiliary Feedwater	N.A.
i. Essential Raw Cooling Water	N.A.
j. CREVS Actuation	N.A.
k. Containment Air Return Fan	N.A.
l. Component Cooling System	N.A.
m. Start Diesel Generators	N.A.
n. Reactor Trip	N.A.
2. Containment Pressure-High	
a. Safety Injection (ECCS)	≤ 27[4]/32[14]/37[5]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8[3]
3) Containment Isolation-Phase "A"[6]	≤ 12[2]/22[1]
4) Containment Ventilation Isolation	≤ 5.0[2][11]
5) Auxiliary Feedwater Pumps	≤ 60[10]
6) Essential Raw Cooling Water	≤ 47[2]/57[1]
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50[2]/60[1]
9) Start Diesel Generators	≤ 12[12]
3. Pressurizer Pressure-Low	
a. Safety Injection (ECCS)	≤ 27[4]/32[14]/37[5]
1) Reactor Trip	≤ 2
2) Feedwater Isolation	≤ 8[3]
3) Containment Isolation-Phase "A"[6]	≤ 12[2]/22[1]
4) Containment Ventilation Isolation	≤ 5.0[2][11]
5) Auxiliary Feedwater Pumps	≤ 60[10]

(continued)

Table 3.3.2-1 (Page 2 of 5)

Engineered Safety Features Actuation System Response Times

INITIATING SIGNAL AND FUNCTION	RESPONSE TIME IN SECONDS
3. Pressurizer Pressure-Low (continued)	
6) Essential Raw Cooling Water	≤ 47[2]/57[1]
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50[2]/60[1]
9) Start Diesel Generators	≤ 12[12]
4. Steam Line Pressure Negative Rate-High	
a. Steam Line Isolation	≤ 8
5. Steam Line Pressure - Low	
a. Safety Injection (ECCS)	≤ 27[4]/32[14]/37[5]
1) Reactor Trip (from SI)	≤ 2
2) Feedwater Isolation	≤ 8[3]
3) Containment Isolation-Phase "A"[6]	≤ 12[2]/22[1]
4) Containment Ventilation Isolation	≤ 5.0[2][11]
5) Auxiliary Feedwater Pumps	≤ 60[10]
6) Essential Raw Cooling Water	≤ 47[2]/57[1]
7) CREVS Actuation	N.A.
8) Component Cooling System	≤ 50[2]/60[1]
9) Start Diesel Generators	≤ 12[12]
b. Steam Line Isolation	≤ 8
6. Containment Pressure - High - High	
a. Containment Spray	≤ 221[13]
b. Containment Isolation-Phase "B"	≤ 68[2]/78[1]
c. Steam Line Isolation	≤ 8
d. Containment Air Return Fans	480 ≤ RT ≤ 600
7. Steam Generator Water Level - High - High	
a. Turbine Trip	≤ 2.5
b. Feedwater Isolation	≤ 8[3]

(continued)

Table 3.3.2-1 (Page 5 of 5)

Engineered Safety Features Actuation System Response Times

TABLE NOTATIONS

(11) Containment purge valves only. Containment radiation monitor valves have a response time of 6.5 seconds.

(12) Diesel generator start time includes a reactor trip response time of 2 seconds.

(13) Includes diesel generator starting, containment spray pump sequence loading-delay/breaker closure, plus stroke time of 1-FCV-72-39/2.

(14) Diesel generator starting and sequence loading delays included. Response time limit includes the opening of valves to establish flowpath and bring pumps to full speed. The additional sequential transfer of ECCS pump suction from the VCT to the RWST (RWST valves open) is included.

(15) Feedwater Isolation Valve (motor) and Feedwater Regulating Valve (air operated) response time includes an ESFAS signal response time of 2 seconds.

TR 3.3 INSTRUMENTATION

TR 3.3.3 Movable Incore Detectors

TR 3.3.3 The Movable Incore Detection System shall be OPERABLE with $\geq$ 75% of the detector thimbles, $\geq$ 2 detector thimbles per core quadrant, and sufficient movable detectors, drive, and readout equipment to map these thimbles.

APPLICABILITY: When the Movable Incore Detection System is used for:

a. Recalibration of the Excore Neutron Flux Detection System, or

b. Monitoring the QUADRANT POWER TILT RATIO, or

c. Measurement of $F^N_{\Delta H}$, $F_Q(Z)$ and F_{XY}.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Movable Incore Detection System inoperable.	A.1 ---------NOTE---------- TR 3.0.3 is not applicable. ----------------------- Restore the inoperable system to OPERABLE status.	Prior to using the system for monitoring or calibration functions.

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. -------NOTE---------- All Required Actions must be completed whenever this Condition is entered. --------------------			
One or more seismic monitoring instruments actuated during a seismic event.	B.1	Restore each actuated monitoring instrument to OPERABLE status.	24 hours
	<u>AND</u>		
	B.2	Perform a CHANNEL CALIBRATION on each actuated monitoring instrument.	10 days
	<u>AND</u>		
	B.3	Analyze data retrieved from actuated instruments to determine the magnitude of the vibratory ground motion.	14 days
	<u>AND</u>		
	B.4	Prepare a report to the NRC in accordance with 10 CFR 50.4 describing the magnitude, frequency spectrum and resultant effect upon facility features important to safety.	14 days

TECHNICAL SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.4-1 to determine which Technical Surveillance Requirements apply for each seismic monitoring instrument.

SURVEILLANCE		FREQUENCY
TSR 3.3.4.1	Perform CHANNEL CHECK.	31 days
TSR 3.3.4.2	Perform CHANNEL OPERATIONAL TEST.	184 days
TSR 3.3.4.3	Perform CHANNEL CALIBRATION.	18 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One reheat stop valve or one reheat intercept valve per low pressure turbine steam line inoperable.	B.1	Restore inoperable valve(s) to OPERABLE status.	72 hours
	<u>OR</u>		
	B.2	Close at least one valve in the affected steam line(s).	78 hours
	<u>OR</u>		
	B.3	Isolate the turbine from the steam supply.	78 hours
C. Turbine Overspeed Protection System inoperable for causes other than Condition A or Condition B.	C.1	Isolate the turbine from the steam supply system.	6 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TR 3.3.5.1	------------------NOTE------------------ TSR 3.0.4 is not applicable. -- Verify the Turbine Overspeed Protection System is OPERABLE in accordance with the plant administrative instruction, "Turbine Integrity Program With Turbine Overspeed Protection (TIPTOP)".	In accordance with TIPTOP.

TR 3.3 INSTRUMENTATION

TR 3.3.6 Loose-Part Detection System

TR 3.3.6 The Loose-Part Detection System shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

---NOTE---
TR 3.0.3 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels of Loose-Part Detection System inoperable > 30 days.	A.1 Prepare and submit a report to the NRC in accordance with 10 CFR 50.4 outlining the cause of the malfunction and the plans for restoring the channel(s) to OPERABLE status.	10 days

TR 3.4 REACTOR COOLANT SYSTEM (RCS)

TR 3.4.1 Safety Valves, Shutdown

TR 3.4.1 One pressurizer Code safety valve shall be OPERABLE with a lift setting of $\geq$ 2410 psig and $\leq$ 2560 psig.

---------------------------NOTE---------------------------
The lift setting is not required to be within the TR limit during MODES 4 and 5 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for entry and operation into and exit from MODES 4 and 5 provided a preliminary cold setting was made prior to heatup.

APPLICABILITY: MODES 4 and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. No pressurizer Code safety valve OPERABLE.	A.1 Suspend all operations involving positive reactivity changes.	Immediately
	<u>AND</u>	
	A.2 Place an OPERABLE Residual Heat Removal (RHR) loop into operation in the shutdown cooling mode.	Immediately

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.1.1	Verify the required pressurizer safety valve OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be within ± 1% of the nominal lift setting of 2485 psig.	In accordance with the Inservice Testing Program.

TR 3.4 REACTOR COOLANT SYSTEM (RCS)

TR 3.4.2 Pressurizer Temperature Limits

TR 3.4.2 The pressurizer temperature shall be limited to:

a. Heatup of ≤ 100°F in any 1-hour period, and

b. Cooldown of ≤ 200°F in any 1-hour period.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. --------NOTE--------- All Required Actions must be completed whenever this Condition is entered. ---------------------		
Pressurizer temperature not within limits.	A.1 Restore pressurizer temperature to within limits.	30 minutes
	<u>AND</u>	
	A.2 Perform engineering evaluation to determine effects of the out-of-limit condition on the structural integrity of the pressurizer.	72 hours
	<u>AND</u>	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.3	Determine that the pressurizer remains acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Reduce pressurizer pressure to < 500 psig.	36 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.2.1	------------------NOTE------------------ Only required during system heatup or cooldown operations. -- Determine that pressurizer temperatures are within limits.	30 minutes.

TR 3.4 REACTOR COOLANT SYSTEM (RCS)

TR 3.4.3 RCS Vents

TR 3.4.3 Two Reactor Coolant System Vent (RCSV) paths shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCSV path inoperable.	A.1 Initiate action to maintain the affected RCSV path closed with power removed from the valve actuators.	Immediately
	<u>AND</u>	
	A.2 Restore the inoperable path to OPERABLE status.	30 days
B. Two RCSV paths inoperable.	B.1 Restore one RCSV path to OPERABLE status.	72 hours
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2 Be in MODE 5.	36 hours

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.3.1	Verify that the upstream manual RCSV isolation valve is locked in the opened position.	18 months
TSR 3.4.3.2	Operate each remotely controlled valve through at least one complete cycle of full travel from the control room.	In accordance with the Inservice Testing Program
TSR 3.4.3.3	Verify flow through the RCSV paths during venting.	18 months

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. (continued)	C.3 Determine that the RCS remains acceptable for continued operation.	Prior to increasing the pressurizer pressure > 500 psig <u>OR</u> Prior to entry to MODE 4.

TECHNICAL SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
TSR 3.4.4.1	-----------------NOTE----------------- Not required with $T_{avg} \leq 250°F$. -------------------------------------- Demonstrate by analysis that RCS dissolved oxygen concentration is $\leq$ 0.10 ppm.	72 hours
TSR 3.4.4.2	Demonstrate by analysis that RCS chloride concentration is $\leq$ 0.15 ppm.	72 hours
TSR 3.4.4.3	Demonstrate by analysis that RCS fluoride concentration is $\leq$ 0.15 ppm.	72 hours

TR 3.4 REACTOR COOLANT SYSTEM (RCS)

TR 3.4.5 Piping System Structural Integrity

TR 3.4.5 The structural integrity of ASME Code Class 1, 2, and 3 components shall be maintained in accordance with TSR 3.4.5.1 and TSR 3.4.5.2.

APPLICABILITY: All MODES.

---------------------------NOTE---------------------------
TR 3.0.4 is not applicable.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Structural integrity of any ASME Code Class 1 component(s) not within limits.	A.1 Restore structural integrity of affected component(s) to within limit. OR	Prior to increasing Reactor Coolant System temperature > 50°F above the minimum temperature required by NDT consider-ations. (continued)

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.3.3	Perform a functional test on a representative sample of snubbers in accordance with Table 3.7.3-4 to determine acceptance with criteria in Table 3.7.3-5.	Each refueling outage.
TSR 3.7.3.4	------------------NOTES---------------- 1. The maximum expected service life for various seals, springs, and other critical parts shall be determined and established based on engineering information and shall be extended or shortened based on monitored test results and failure history. 2. Critical parts shall be replaced so that the maximum service life will not be exceeded during a period when the snubber is required to be OPERABLE. 3. The parts replacement shall be documented and the documentation shall be retained for the duration of the unit operating license. -------------------------------------- Verify that the service life of hydraulic and mechanical snubbers has not been exceeded or will not be exceeded prior to the next scheduled surveillance inspection.	Each refueling outage.

Table 3.7.3-1 (Page 1 of 1)

Snubber Visual Inspection Acceptance Criteria

---

1. Visual inspection shall verify that:

   a. There are no visible indications of damage or impaired OPERABILITY.

   b. Attachments to the foundation or supporting structure are functional; and

   c. Fasteners for attachment of the snubber to the component and to the snubber anchorage are functional.

2. Snubbers which appear inoperable as a result of visual inspections shall be classified as unacceptable and may be reclassified acceptable for the purpose of establishing the next visual inspection interval, provided that:

   a. The cause of the rejection is clearly established and remedied for that particular snubber and for other snubbers irrespective of type that may be generically susceptible; and

   b. The affected snubber is functionally tested in the as-found condition and determined OPERABLE per Table 3.7.3-5, Snubber Functional Test Acceptance Criteria.

3. All snubbers connected to an inoperable common hydraulic fluid reservoir shall be counted as unacceptable for determining the next inspection interval.

4. Snubbers which have been made inoperable as the result of unexpected transients, isolated damage or other such random events, when the provisions of Table 3.7.3-3 have been met and any other appropriate corrective action implemented, shall not be counted in determining the next visual inspection interval.

5. A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the ACTION requirements shall be met.

Table 3.7.3-2 (Page 1 of 2)

Snubber Visual Inspection Surveillance Frequency

Population or Category (Notes 1 and 2)	NUMBER OF UNACCEPTABLE SNUBBERS		
	Column A Extended Interval (Notes 3 and 6)	Column B Repeat Interval (Notes 4 and 6)	Column C Reduce Interval (Notes 5 and 6)
1	0	0	1
80	0	0	2
100	0	1	4
150	0	3	8
200	2	5	13
300	5	12	25
400	8	18	36
500	12	24	48
750	20	40	78
1000 or greater	29	56	109

Note 1: The next visual inspection interval for a snubber population or category size shall be determined based upon the previous inspection interval and the number of unacceptable snubbers found during that interval. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly. However, the licensee must make and document that decision before any inspection and shall use that decision as the basis upon which to determine the next inspection interval for that category.

Note 2: Interpolation between population or category size and the number of unacceptable snubbers is permissible. Use next lower integer for the value of the limit for Columns A, B, or C if that integer includes a fractional value of unacceptable snubbers as described by interpolation.

(continued)

Table 3.7.3-2 (Page 2 of 2)

Snubber Visual Inspection Surveillance Frequency

Note 3: If the number of unacceptable snubbers is equal to or less than the number in Column A, the next inspection interval may be twice the previous interval but not greater than 48 months.

Note 4: If the number unacceptable snubbers is equal to or less than the number in Column B but greater than the number in Column A, the next inspection interval shall be the same as the previous interval.

Note 5: If the number of unacceptable snubbers is equal to or greater than the number in Column C, the next inspection interval shall be two-thirds of the previous interval. However, if the number of unacceptable snubbers is less than the number in Column C but greater than the number in Column B, the next interval shall be reduced proportionally by interpolation, that is, the previous interval shall be reduced by a factor that is one third of the ratio of the difference between the number of unacceptable snubbers found during the previous interval and the number in Column B to the difference in the numbers in Column B and C.

Note 6: The provisions of TSR 3.0.2 are applicable for all inspection intervals up to and including 48 months.

Table 3.7.3-3 (Page 1 of 1)

Snubber Transient Event Inspection

1. An inspection shall be performed of all hydraulic and mechanical snubbers attached to sections of systems that have experienced unexpected, potentially damaging transients as determined from a review of operational data and a visual inspection of the systems within six months following such an event.

2. In addition to satisfying the visual inspection acceptance criteria, freedom-of-motion of mechanical snubbers shall be verified using one of the following:

    a. Manually induced snubber movement.

    b. Evaluation of in-place snubber piston setting.

    c. Stroking the mechanical snubber through its full range of travel.

Table 3.7.3-4 (Page 1 of 2)

Snubber Functional Testing Plan

1. The representative sample of snubbers shall include each type and shall be tested using sample plan A for hydraulic snubbers and sample plan B for mechanical snubbers.

2. The NRC Regional Administrator shall be notified in writing of any changes to the sample plan prior to the test period.

SAMPLE PLAN A

1. At least 10% of the total hydraulic snubber population shall be functionally tested either in-place or in a bench test.

2. For each hydraulic snubber of a type that does not meet the functional test acceptance criteria of Table 3.7.3-5, an additional 10% of hydraulic snubbers shall be functionally tested until no more failures are found or until all hydraulic snubbers have been functionally tested.

SAMPLE PLAN B

1. An initial representative sample of 37 mechanical snubbers shall be functionally tested in accordance with Figure 3.7.3-1. For each mechanical snubber type which does not meet the functional test acceptance criteria of Table 3.7.3-5, another sample of at least 19 snubbers shall be tested. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 36.49 + 18.18C$. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls in the "Accept" region or all mechanical snubbers have been tested.

(continued)

Table 3.7.3-4 (Page 2 of 2)

Snubber Functional Testing Plan

TABLE NOTES

1. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time provided all snubbers tested with the failed equipment during the day of equipment failure are retested.

2. The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure, as far as practicable, that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type.

3. Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan.

4. If during the functional testing, additional sampling is required due to failure of only one type of snubber, the functional test results shall be reviewed at that time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

Table 3.7.3-5 (Page 1 of 1)

Snubber Functional Testing Acceptance Criteria

The snubber functional test shall verify that:

a. Activation (restraining action) is achieved within the specified range in both tension and compression.

b. Snubber bleed, or release rate where required, is present in both tension and compression, within the specified range (Hydraulic Snubbers).

c. The force required to initiate or maintain motion of the snubber is within the specified range in both directions of travel (Mechanical Snubbers).

TABLE NOTES

1. Testing methods may be used to measure parameters indirectly or parameters other than those specified if those results can be correlated to the specified parameters through established methods.

2. An engineering evaluation shall be made of each failure to meet the functional test criteria to determine the cause of the failure. The results of this evaluation shall be used, if applicable, in selecting snubbers to be tested in an effort to determine the OPERABILITY of other snubbers irrespective of the type which may be subject to the same failure mode.

3. For snubbers found inoperable, an engineering evaluation shall be performed on the components to which the inoperable snubbers are attached. The purpose of this engineering evaluation shall be to determine if the components to which the inoperable snubbers are attached were adversely affected by the inoperability of the snubbers in order to ensure that the component remains capable of meeting the designed service.

4. If any snubber selected for functional testing either fails to lock up or fails to move, i.e., frozen-in-place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same type subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated in Table 3.7.3-4 for snubbers not meeting the functional test acceptance criteria.

TECHNICAL SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
TSR 3.7.4.4	-----------------NOTE----------------- Only applicable to excore fission detectors. -------------------------------------- Determine that the removable contamination is < 0.005 microcuries for each excore fission detector.	Within 31 days prior to being installed. <u>AND</u> Following repair or maintenance to the source.

TR 3.7 PLANT SYSTEMS

TR 3.7.5 Area Temperature Monitoring

TR 3.7.5 The normal temperature limit of each area shown in Table 3.7.5-1 shall not be exceeded for > 8 hours and the abnormal temperature limits shall not be exceeded.

APPLICABILITY: Whenever the affected equipment in an area is required to be OPERABLE.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more areas exceeding normal temperature limits for > 8 hours.	A.1 ----------NOTE---------- TR 3.0.3 and TR 3.0.4 are not applicable. ------------------------ Prepare and submit to the NRC a report in accordance with 10 CFR 50.4 that provides a record of the cumulative time and the amount by which the temperature in the affected area(s) exceeded the limit(s) and an analysis to demonstrate OPERABILITY of the affected equipment.	30 days

(continued)

BASES

TR (continued)	b. A flow path from an OPERABLE RWST through a charging pump to the RCS.
APPLICABILITY	The OPERABILITY of one boron injection flow path ensures that this system is available for reactivity control while in MODES 4, 5, and 6. Boron injection flow paths for MODES 1, 2, and 3 are covered in Technical Requirement 3.1.2, "Boration Systems Flow Paths, Operating".
ACTIONS	<u>A.1 and A.2</u> With the Boration Systems flow path OPERABILITY requirements not met, or the Boration Systems flow path not capable of being powered by an OPERABLE emergency power source, the plant must be placed in a condition where negative reactivity addition is not required. This is accomplished by suspending all CORE ALTERATIONS and positive reactivity additions immediately. One boron injection flow path is required to meet the TR and to ensure that negative reactivity control is available during MODES 4, 5, and 6.. Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. The immediate Completion Time is consistent with the required times for actions to be performed without delay and in a controlled manner.
TECHNICAL SURVEILLANCE REQUIREMENTS	<u>TSR 3.1.1.1</u> This surveillance verifies the temperature of the areas containing portions of the flow path from the boric acid tanks is $\geq$ 63°F. This ensures that the high concentration of boric acid in the storage tanks is not allowed to precipitate due to cooling. The Surveillance is modified by a note stating that the surveillance is required only if a flow path from the boric

(continued)

BASES

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.1.1.1 (continued)

acid storage tanks is required OPERABLE. The 12 hour Frequency is based on engineering experience and is reasonable considering the time required for performing the surveillance and the probability for changes in the area temperatures.

TSR 3.1.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the Boration System flow path provides assurance that the proper flow paths exist for Boration System operation. This TSR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to being locked, sealed, or secured. This TSR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position. This TSR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

REFERENCES

1. Watts Bar FSAR, Section 9.3.4, "Chemical and Volume Control System."

2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

3. CEN-603, "Boric Acid Concentration Reduction Effort, Technical Bases and Operational Analysis for Watts Bar Nuclear Plant, Unit 1, Revision 00, April 1993.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Safety Valves, Shutdown

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Trip System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop-type, spring-loaded, self-actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2735 psig, which is 110% of the design pressure (Ref. 1).

Because the safety valves are totally enclosed and self-actuating, they are considered independent components. The relief capacity for each valve, 420,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, 5, and 6 (with the reactor vessel head on); however, in MODE 4, MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of Technical Specification 3.4.12, "Cold Overpressure Mitigation System (COMS)."

The upper and lower pressure limits are based on the ± 3% tolerance. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents above 350 °F. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of

(continued)

BASES

BACKGROUND (continued)	design pressure. The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 2) could include damage to RCS components, increased LEAKAGE, or a requirement to perform additional stress analyses prior to resumption of reactor operation.
APPLICABLE SAFETY ANALYSES	The pressurizer safety valves protect the RCS from being pressurized above the RCS pressure Safety Limit. The pressurizer safety valves provide overpressurization protection during both power operation and hot standby. However, the pressurizer safety valves are not assumed to function to mitigate a DBA or transient in MODES 4 and 5 (Ref. 3).
TR	This requirement is provided to ensure continuity in the restructuring of Standard Technical Specifications. Reactor Coolant System overpressure protection is provided in MODES 4 and 5 by the Cold Overpressure Mitigation System (COMS) covered by LCO 3.4.12. Reference 4 specifies requirements which, when met, may preclude the need for this TR. A Note modifies this TR to indicate that the lift setting of the pressurizer code safety valves can be outside the required lift setting when in MODE 4 for the purpose of setting at hot ambient conditions. Safety valves can lift at a slightly different pressure as the valve temperature vary. Therefore, setting the safety valve for nominal operating conditions in MODE 1 may result in a lift pressure drifting outside the required tolerance limits as the plant is shutdown to MODE 5. This exception is allowed for entry and operation into and exit from MODES 4 and 5 provided a preliminary cold setting was made prior to heatup.
APPLICABILITY	The OPERABILITY of one pressurizer Code safety valve ensures that overpressure protection is provided in MODES 4 and 5. OPERABILITY of Code safety valves is not required in MODE 6. Code safety valve OPERABILITY requirements for MODES 1, 2, and 3 are covered in Technical Specification 3.4.10, "Pressurizer Safety Valves."

(continued)

BASES (continued)

ACTIONS

A.1

With no pressurizer safety valves OPERABLE, the plant must be placed in a condition which minimizes the risk of a pressure spike large enough to actuate a safety valve. This is done by suspending all operations involving positive reactivity changes. The immediate Completion Time for performance of Required Action A.1 shall not preclude completion of actions to establish a safe condition.

A.2

In addition to Action A.1, an OPERABLE Residual Heat Removal loop shall be placed in operation in the shutdown cooling mode. This provides overpressure protection through the Residual Heat Removal suction and discharge relief valves. The immediate Completion Time requires an operator to initiate actions to place the loop in shutdown cooling. Once actions are initiated, they must be continued until the loop is in the shutdown cooling mode.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.1.1

TSR 3.4.1.1 requires verification that the pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program.

REFERENCES

1. Watts Bar FSAR, Section 5.5.13, "Safety and Relief Valves."

2. ASME Boiler and Pressure Vessel Code, Section III, NB 7000.

3. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

4. Generic Letter 90-06, "Resolution of Generic Issue 70, "Power-Operated Relief Valve and Block Valve Reliability," and Generic Issue 94, "Additional Low-Temperature Overpressure Protection for Light-Water Reactors," Pursuant to 10 CFR 50.54(f).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Pressurizer Temperature Limits

BASES

BACKGROUND

The pressurizer is an ASME Section III, vertical vessel with hemispherical top and bottom heads constructed of carbon steel. The vessel is clad with austenitic stainless steel on all surfaces exposed to the reactor coolant. A stainless steel liner or tube may be used in lieu of cladding in some nozzles. The surge line nozzle and removable electric heaters are installed in the bottom head. Spray line nozzles, relief and safety valves are located in the top head of the vessel. A small continuous spray is provided through a manual bypass valve around the power-operated spray valves. The temperature, and hence the pressure are controlled by varying the power input to selected heater elements. The pressurizer is designed to withstand the effects of cyclic loads due to pressure and temperature changes. These loads are introduced by startup and shutdown operations, power transients and reactor trips. During startup and shutdown, the rate of temperature change is controlled by the operator. Heatup rate is controlled by the input to the heater elements, and cooldown is controlled by spray. When the pressurizer is filled with water, i.e., during initial system heatup, and near the end of the second phase of plant cooldown, Reactor Coolant System (RCS) pressure is maintained by the letdown flow rate via the Residual Heat Removal System.

These Bases address the control of the rate of change of temperature and the effect of the thermal cycling on critical areas of the pressure boundary of the pressurizer. The Reactor Coolant Pressure Boundary, which includes the pressurizer, is defined in 10 CFR 50, section 50.2 (Ref.1). General rules for design and fabrication are provided in 10 CFR 50, section 50.55a (Ref. 2). These design and fabrication rules are based on the ASME Boiler and Pressure Vessel Code.

APPLICABLE SAFETY ANALYSES

The limits on the rate of change of temperature for the heatup and cooldown of the pressurizer are not derived from Design Basis Accident analyses (Ref. 3). The limits are prescribed during normal operation to limit the cyclic, thermal loading on critical areas in the pressure

(continued)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Vents

BASES

BACKGROUND

The Reactor Vessel Head Vent System (RVHVS) is installed on the reactor vessel head. The RVHVS consists of a safety-grade venting flow path with redundancy around process solenoid valves. Two one inch solenoid-operated globe valves are mounted in series in each redundant portion of the flow path. The piping between these valves is provided with a temperature monitor. Any leakage through the upstream valve will be detected as an increase in temperature. The two redundant upstream valves are open/close isolation valves and are powered by opposite vital power buses. The two redundant downstream valves are throttle valves that are used to regulate the release rate of the noncondensible gases and steam. The two throttle valves are also powered by opposite vital power buses. All four valves are remote, manual-operated from the control room. The valves are normally closed, deenergized and designed to fail closed in accordance with Regulatory Guide 1.48. The system provides venting during plant startup/shutdown or for postaccident. The system is designed to operate in the containment atmosphere during and after a design basis event. However, the system is not utilized during emergency operation until an inadequate water level in the reactor vessel has been determined. During an incident with hydrogen generation and release, a venting period of approximately ten minutes is acceptable without violating the combustible concentration of hydrogen in the containment.

The capability and the function of the system is consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements" (Ref.1). Direct operator action is required to actuate the system. System actuation is only required when the accumulation of noncondensible gases could impair forced or natural circulation and, hence, cooling of the core.

APPLICABLE SAFETY ANALYSIS

The RVHVS is designed to ensure that noncondensible gases do not accumulate under the reactor vessel head and thereby impair the cooling of the core. However, in designing the accident sequences for theoretical hazard evaluation, the RVHVS is not assumed to be a system that

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

directly serves to prevent or mitigate a DBA or transient (Ref. 2).

TR

TR 3.4.3 requires that the two redundant vent paths are OPERABLE. One condition for OPERABILITY is that the upstream manual isolation valve is locked open. However, in case of one inoperable path, one condition for continued operation, (while restorative actions take place), is that the inoperable path is maintained closed with power removed from both valve actuators. With two paths inoperable, no requirement exists with respect to isolation during the much shorter time of restorative actions.

APPLICABILITY

The TR is basically protecting against uncovering the core and reduces the possibility for impairment of natural or forced circulation through the core. This is mainly a concern during the production of power and early in the decay heat removal phase. Accordingly, Applicability is consistent with operation in MODES 1, 2, 3 and 4. In higher-numbered MODES, the heat flux in the core is low and protection by this TR is not required.

ACTIONS

A.1 and A.2

With one vent path inoperable, it is necessary to immediately start actions to see to that the inoperable path is closed and fully isolated from the Reactor Coolant System. The inoperable path must be restored to OPERABLE condition in 30 days. It should be noted that during this period of time one path is fully OPERABLE. If the need for venting should occur during this time period, the OPERABLE path will provide 100% of the required venting capacity. Based on this, 30 days is an acceptable time period for restoring the inoperable path.

B.1

With two paths inoperable, it is required to restore one path in 72 hours. The 72 hours is based on operating experience and is a reasonable time period for identifying

(continued)

BASES

ACTIONS

B.1 (continued)

and correcting problems which could be associated with an inoperable path.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A or B are not met, the plant must be placed in a condition in which the TR does not apply. This is accomplished by placing the unit in MODE 3 within 6 hours and MODE 5 in an additional 30 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging plant systems.

TECHNICAL SURVEILLANCE REQUIREMENTS

TSR 3.4.3.1, TSR 3.4.3.2 and TSR 3.4.3.3

Every 18 months it is necessary to verify that each of the two vent paths are OPERABLE. This verification consists of checking the upstream isolation valve and ensuring that the valve is locked in the open position. Further, the two control valves are operated from the control room, in accordance with the Inservice Testing Program through one complete cycle of full travel. Lastly, the test includes a verification of flow through the two vent paths.

REFERENCES

1. NUREG-0737, "Clarification of TMI Action Plan Requirements."
2. WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 Chemistry

BASES

BACKGROUND

The Reactor Coolant System (RCS) water chemistry is selected to minimize corrosion. A periodic analysis of the coolant chemical composition is performed to verify that the reactor coolant quality meets the specifications (Ref. 1). This Technical Requirement places limits on the dissolved oxygen, chloride and fluoride content of the RCS to minimize corrosion.

Limiting dissolved oxygen content of the RCS limits the amount of general corrosion and reduces the possibility of stress corrosion. General corrosion is a contributing factor in Reactor Coolant Activity (Ref. 2) and must be controlled for ALARA (as low as reasonably achievable) considerations as well as structural integrity considerations.

Both chlorides and fluorides have been shown to cause stress corrosion if present in the RCS in sufficiently high concentrations at high pressure and temperature conditions. Stress corrosion can lead to either localized leakage or catastrophic failure of the RCS. The associated effects of exceeding the dissolved oxygen, chloride, and fluoride limits are time and temperature dependent. Corrosion studies show that operation may be continued with contaminant concentration levels in excess of the Steady-State Limits, up to the Transient Limits, for the specified limited time intervals without having a significant effect on the structural integrity of the RCS.

APPLICABLE SAFETY ANALYSES

Minimizing corrosion of the RCS reduces the potential for RCS leakage and for failure due to stress corrosion, thus ultimately ensuring the structural integrity of the RCS (Ref. 3). It is not, however, a consideration in the analyses of Design Basis Accidents.

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BASES (continued)

REFERENCES	1.	Watts Bar FSAR, Section 5.2, "Integrity of Reactor Coolant Pressure Boundary."
	2.	Watts Bar FSAR, Section 11.1, "Source Terms."
	3.	WCAP-11618, "MERITS Program-Phase II, Task 5, Criteria Application," including Addendum 1 dated April, 1989.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 Piping System Structural Integrity

BASES

BACKGROUND

Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves are performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code (Ref. 1) and applicable Addenda, as required by 10 CFR 50.55a(g) (Ref.2). Exception to these requirements apply where relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i) and (a)(3). In general, the surveillance intervals specified in Section XI of the ASME Code apply. However, the Inservice Inspection Program includes a clarification of the frequencies for performing the inservice inspection and testing activities required by Section XI of the ASME Code. This clarification is provided to ensure consistency in surveillance intervals throughout the Technical Specifications. Each reactor coolant pump flywheel is, in addition, inspected as recommended in Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975 (Ref.3).

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Agenda. Accordingly, in order to establish proper test conditions without MODE limitations, the Specification takes an exception to the provisions in TR 3.0.4. This provides the necessary authorization to change the operational condition, including change of MODE, in order to establish the proper test conditions as stipulated in the ASME Boiler and Pressure Vessel Code.

Additionally, programmatic information on Inservice Inspection is provided in Technical Specifications, Chapter 5.0, Administrative Controls, Section 5.7.2.11, Inservice Inspection Program.

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