Duke Power
A Duke Energy Company

OCONEE NUCLEAR STATION
IMPROVED TECHNICAL SPECIFICATIONS

VOLUME 3

SECTION
3.3
ITS
ITS Bases
CTS Markup
NSHC

DUKE POWER
Generating Excellence
Oconee Nuclear Site

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.3 - INSTRUMENTATION

ATTACHMENT 1

TECHNICAL SPECIFICATIONS

3.3 INSTRUMENTATION

3.3.1 Reactor Protective System (RPS) Instrumentation

LCO 3.3.1 Three channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required channel inoperable.	A.1	Place channel in trip.	1 hour
B. Two or more required channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1	Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately
C. As required by Required Action B.1 and referenced in Table 3.3.1-1.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Open all control rod drive (CRD) trip breakers.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. As required by Required Action B.1 and referenced in Table 3.3.1-1.	D.1	Open all CRD trip breakers.	6 hours
E. As required by Required Action B.1 and referenced in Table 3.3.1-1.	E.1	Reduce THERMAL POWER < 30% RTP.	6 hours
F. As required by Required Action B.1 and referenced in Table 3.3.1-1.	F.1	Reduce THERMAL POWER < 2% RTP.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	-------------------NOTE------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare results of calorimetric heat balance calculation to the power range channel output and adjust power range channel output if calorimetric exceeds power range channel output by ≥ 2% RTP.	24 hours
SR 3.3.1.3	-------------------NOTE------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Compare out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE and adjust power range channel output if the absolute difference between the power range and incore measurements is ≥ 2% RTP.	31 days
SR 3.3.1.4	-------------------NOTE------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP. -- Calibrate power range channel output to the calorimetric coincident with the imbalance output being calibrated to the imbalance condition determined by the incore detector system.	31 days
SR 3.3.1.5	Perform CHANNEL FUNCTIONAL TEST.	45 days on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	-------------------NOTE-------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

Table 3.3.1-1 (page 1 of 1)
Reactor Protective System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION B.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower –				
a. High Setpoint	1,2(a),3(d)	C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6	≤ 105.5% RTP
b. Low Setpoint	2(b),3(b) 4(b),5(b)	D	SR 3.3.1.1 SR 3.3.1.6	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	≤ 618°F
3. RCS High Pressure	1,2(a),3(d)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	≤ 2355 psig
4. RCS Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	≥ 1800 psig
5. RCS Variable Low Pressure	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	As specified in the COLR
6. Reactor Building High Pressure	1,2,3(c)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	≤ 4 psig
7. Reactor Coolant Pump to Power	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	>2% RTP with ≤ 2 pumps operating
8. Nuclear Overpower Flux/Flow Imbalance	1,2(a)	C	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6	As specified in the COLR
9. Main Turbine Trip (Hydraulic Fluid Pressure)	≥ 30% RTP	E	SR 3.3.1.5 SR 3.3.1.6	≥ 800 psig
10. Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)	≥ 2% RTP	F	SR 3.3.1.5 SR 3.3.1.6	≥ 75 psig
11. Shutdown Bypass RCS High Pressure	2(b),3(b) 4(b),5(b)	D	SR 3.3.1.1 SR 3.3.1.5 SR 3.3.1.6	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) When not in shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

3.3 INSTRUMENTATION

3.3.2 Reactor Protective System (RPS) Manual Reactor Trip

LCO 3.3.2 The RPS Manual Reactor Trip Function shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Manual Reactor Trip Function inoperable.	A.1 Restore Function to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Open all CRD trip breakers.	12 hours 12 hours
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL FUNCTIONAL TEST.	Once prior to each reactor startup if not performed within the previous 7 days

3.3 INSTRUMENTATION

3.3.3 Reactor Protective System (RPS) – Reactor Trip Module (RTM)

LCO 3.3.3 Four RTMs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RTM inoperable.	A.1.1 Trip the associated CRD trip breaker.	1 hour
	<u>OR</u>	
	A.1.2 Remove power from the associated CRD trip breaker.	1 hour
	<u>AND</u>	
	A.2 Physically remove the inoperable RTM.	1 hour
B. Two or more RTMs inoperable in MODE 1, 2, or 3. <u>OR</u> Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	12 hours
	<u>AND</u>	
	B.2.1 Open all CRD trip breakers.	12 hours
	<u>OR</u>	
	B.2.2 Remove power from all CRD trip breakers.	12 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more RTMs inoperable in MODE 4 or 5. OR Required Action and associated Completion Time not met in MODE 4 or 5.	C.1	Open all CRD trip breakers.	6 hours
	OR		
	C.2	Remove power from all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL FUNCTIONAL TEST.	31 days

3.3 INSTRUMENTATION

3.3.4 Control Rod Drive (CRD) Trip Devices

LCO 3.3.4 The following CRD trip devices shall be OPERABLE:

a. Two AC CRD trip breakers;

b. Two DC CRD trip breaker pairs; and

c. Eight electronic trip assembly (ETA) relays.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each CRD trip device.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more CRD trip breakers or breaker pair diverse trip Functions inoperable.	A.1	Trip the CRD trip breaker.	48 hours
	<u>OR</u>		
	A.2	Remove power from the CRD trip breaker.	48 hours
B. One or more CRD trip breakers or breaker pair inoperable for reasons other than those in Condition A.	B.1	Trip the CRD trip breaker.	1 hour
	<u>OR</u>		
	B.2	Remove power from the CRD trip breaker.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more ETA relays inoperable.	C.1	Transfer affected CONTROL ROD group to power supply with OPERABLE ETA relays.	1 hour
	OR		
	C.2	Trip corresponding AC CRD trip breaker(s).	1 hour
D. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	D.1	Be in MODE 3.	12 hours
	AND		
	D.2.1	Open all CRD trip breakers.	12 hours
	OR		
	D.2.2	Remove power from all CRD trip breakers.	12 hours
E. Required Action and associated Completion Time not met in MODE 4 or 5.	E.1	Open all CRD trip breakers.	6 hours
	OR		
	E.2	Remove power from all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	31 days

3.3 INSTRUMENTATION

3.3.5 Engineered Safeguards Protective System (ESPS) Analog Instrumentation

LCO 3.3.5 Three channels of ESPS analog instrumentation for each Parameter in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each Parameter.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one channel inoperable.	A.1 Place channel in trip.	1 hour
B. One or more Parameters with two or more channels inoperable. OR Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2.1 --------NOTE--------- Only required for RCS Pressure – Low. -------------------- Reduce RCS pressure < 1750 psig. AND	12 hours 36 hours
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2 --------NOTE--------- Only required for RCS Pressure – Low Low. ---------------------	
	Reduce RCS pressure < 900 psig.	36 hours
	AND	
	B.2.3 --------NOTE--------- Only required for Reactor Building Pressure – High and High High. ---------------------	
	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.5.1 Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.5.2	-------------------NOTE-------------------- When an ESPS analog channel is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided the remaining two channels of ESPS analog instrumentation are OPERABLE or tripped. --- Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.5-1 (page 1 of 1)
Engineered Safeguards Protective System Analog Instrumentation

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Reactor Coolant System Pressure – Low	≥ 1750 psig	≥ 1500 psig
2. Reactor Coolant System Pressure – Low Low	≥ 900 psig	≥ 500 psig
3. Reactor Building (RB) Pressure – High	1,2,3,4	≤ 4 psig
4. Reactor Building Pressure – High High	1,2,3,4	≤ 15 psig

3.3 INSTRUMENTATION

3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the ESPS Functions below shall be OPERABLE:

a. High Pressure Injection, Reactor Building (RB) Non-Essential Isolation, Keowee Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input (ES Channels 1 and 2);

b. Low Pressure Injection and RB Essential Isolation (ES Channels 3 and 4);

c. RB Cooling, RB Essential Isolation and Penetration Room Ventilation (ES Channels 5 and 6); and

d. RB Spray (ES Channels 7 and 8).

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguard equipment is required to be OPERABLE.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESPS Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.6.1 Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels

LCO 3.3.7 Eight ESPS Digital Automatic Actuation Logic Channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguard (ES) equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each automatic actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital automatic actuation logic channels inoperable.	A.1 Place associated component(s) in ES configuration.	1 hour
	<u>OR</u>	
	A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital automatic actuation logic CHANNEL FUNCTIONAL TEST.	31 days

3.3 INSTRUMENTATION

3.3.8 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.8 The PAM instrumentation for each Function in Table 3.3.8-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Not applicable to Functions 14, 18, 19, and 20. --------------------- One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.6.	Immediately

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Not applicable to Functions 10, 14, 18, 19, and 20. --------------------- One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days
D. ---------NOTE--------- Only applicable to Function 10. --------------------- Two required channels inoperable.	D.1 Restore one required channel to OPERABLE status.	72 hours
E. ---------NOTE--------- Only applicable to Function 14. --------------------- One required channel inoperable.	E.1 Restore required channel to OPERABLE status.	24 hours
F. ---------NOTE--------- Only applicable to Functions 18, 19, and 20. --------------------- One or more Functions with required channel inoperable.	F.1 Declare the affected train inoperable.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
G. Required Action and associated Completion Time of Condition C, D or E not met.	G.1	Enter the Condition referenced in Table 3.3.8-1 for the channel.	Immediately
H. As required by Required Action G.1 and referenced in Table 3.3.8-1.	H.1	Be in MODE 3.	12 hours
	AND		
	H.2	Be in MODE 4.	18 hours
I. As required by Required Action G.1 and referenced in Table 3.3.8-1.	I.1	Initiate action in accordance with Specification 5.6.6.	Immediately

SURVEILLANCE REQUIREMENTS

-----------------------------------NOTE-----------------------------------
These SRs apply to each PAM instrumentation Function in Table 3.3.8-1 except where indicated.

SURVEILLANCE		FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.8.2	------------------NOTE------------------ Only applicable to PAM Functions 7 and 10. -- Perform CHANNEL CALIBRATION.	12 months
SR 3.3.8.3	------------------NOTE------------------ 1. Neutron detectors are excluded from CHANNEL CALIBRATION. 2. Not applicable to PAM Functions 7 and 10. -- Perform CHANNEL CALIBRATION.	18 months

Table 3.3.8-1 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION G.1
1.	Wide Range Neutron Flux	2	H
2.	RCS Hot Leg Temperature	2 per loop	H
3.	RCS Hot Leg Level	2	I
4.	RCS Pressure (Wide Range)	2	H
5.	Reactor Vessel Head Level	2	I
6.	Containment Sump Water Level (Wide Range)	2	H
7.	Containment Pressure (Wide Range)	2	H
8.	Containment Isolation Valve Position	2 per penetration flow path[a][b][c]	H
9.	Containment Area Radiation (High Range)	2	I
10.	Containment Hydrogen Concentration	2	H
11.	Pressurizer Level	2	H
12.	Steam Generator Water Level	2 per SG	H
13.	Steam Generator Pressure	2 per SG	H
14.	Borated Water Storage Tank Water Level	2	H
15.	Upper Surge Tank Level	2	H
16.	Core Exit Temperature	2 independent sets of 5[d]	H
17.	Subcooling Monitor	2	H
18.	HPI System Flow	1 per train	NA
19.	LPI System Flow	1 per train	NA
20.	Reactor Building Spray Flow	1 per train	NA
21.	Emergency Feedwater Flow	2 per SG	H

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured.

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel.

(c) Position indication requirements apply only to containment isolation valves that are electrically controlled.

(d) The subcooling margin monitor takes the average of the five highest CETs for each of the ICCM trains.

3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux

LCO 3.3.9 Two source range neutron flux channels shall be OPERABLE.

APPLICABILITY: MODES 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required source range neutron flux channel inoperable with THERMAL POWER level ≤ 4E-4% RTP on the wide range neutron flux channels.	A.1	Restore channel to OPERABLE status.	Prior to increasing THERMAL POWER
B. Two required source range neutron flux channels inoperable with THERMAL POWER level ≤ 4E-4% RTP on the wide range neutron flux channels.	B.1	Suspend operations involving positive reactivity changes.	Immediately
	AND		
	B.2	Initiate action to insert all CONTROL RODS.	Immediately
	AND		
	B.3	Open CONTROL ROD drive trip breakers.	1 hour
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Verify SDM to be within the limit specified in the COLR.	1 hour AND Once per 12 hours thereafter
C. One or more source range neutron flux channel(s) inoperable with THERMAL POWER level > 4E-4% RTP on the wide range neutron flux channels.	C.1 Initiate action to restore affected channel(s) to OPERABLE status.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	-------------------NOTE-------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.10 Wide Range Neutron Flux

LCO 3.3.10 Two wide range neutron flux channels shall be OPERABLE.

APPLICABILITY: MODE 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required channel inoperable.	A.1 Reduce THERMAL POWER to < 4E-4% RTP.	2 hours
B. Two required channels inoperable.	B.1 Suspend operations involving positive reactivity changes. <u>AND</u> B.2 Open CRD trip breakers.	Immediately 1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.10.1 Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.10.2	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.10.3	Verify at least one decade overlap between source range and wide range neutron flux channels.	Once each reactor startup prior to the source range indication exceeding 10^5 cps if not performed within the previous 7 days

3.3 INSTRUMENTATION

3.3.11 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Instrumentation

LCO 3.3.11 Three MSLB Detection and MFW Isolation instrumentation channels per steam generator (SG) shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with main steam header pressure $\geq$ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each SG (MFW Isolation Function).

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more MFW Isolation Functions with one channel inoperable.	A.1 Place channel(s) in trip.	4 hours
B. One or more MFW Isolation Functions with two or more channels inoperable. <u>OR</u> Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2.1 Reduce main steam header pressure to < 700 psig. <u>OR</u> B.2.2 Close all MFCVs and SFCVs.	12 hours 18 hours 18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.11.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.11.2 Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.12 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Manual Initiation

LCO 3.3.12 Two MSLB Detection and MFW Isolation manual initiation switches shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with main steam header pressure $\geq$ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One manual initiation switch inoperable.	A.1	Restore manual initiation switch to OPERABLE status.	72 hours
B Two manual initiation switches inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2.1	Reduce main steam header pressure to < 700 psig.	18 hours
	OR		
	B.2.2	Close all MFCVs and SFCVs.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.12.1 Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.13 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Logic Channels

LCO 3.3.13 Two MSLB Detection and MFW Isolation Logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with main steam header pressure $\geq$ 700 psig
except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One logic channel inoperable.	A.1	Restore channel to OPERABLE status.	72 hours
B. Two logic channels inoperable. OR Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2.1	Reduce main steam header pressure to < 700 psig.	18 hours
	OR		
	B.2.2	Close all MFCVs and SFCVs.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.13.1 Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.14 Emergency Feedwater (EFW) Pump Initiation Circuitry

LCO 3.3.14 Two loss of main feedwater (LOMF) pump instrumentation channels for each automatic initiation circuit, and an automatic and manual initiation circuit for each EFW pump shall be OPERABLE.

-------------------------NOTE-------------------------
The EFW pump automatic initiation circuit is not required to be OPERABLE in MODES 3 and 4.

APPLICABILITY: MODES 1, 2 and 3,
MODE 4 when the steam generator is relied upon for heat removal.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each EFW pump initiation circuit.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more EFW pump automatic initiation circuits with one LOMF channel inoperable.	A.1	Place channel(s) in trip.	1 hour
B. One or more required EFW pump initiation circuits inoperable. OR Required Action and associated Completion Time not met.	B.1	Declare the affected EFW pump(s) inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.14.1 Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.14.2 Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.15 Turbine Stop Valve (TSV) Closure

LCO 3.3.15 Two TSV Closure channels shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3 except when all TSVs are closed.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more TSV Closure channel(s) inoperable.	A.1 Declare the TSVs inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.15.1 Perform CHANNEL FUNCTIONAL TEST.	31 days

3.3 INSTRUMENTATION

3.3.16 Reactor Building (RB) Purge Isolation—High Radiation

LCO 3.3.16 One channel of Reactor Building Purge Isolation—High Radiation shall be OPERABLE.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within the containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place and maintain RB purge valves in closed positions.	Immediately
	<u>OR</u>	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	<u>AND</u>	
	A.2.2 Suspend movement of irradiated fuel assemblies within the containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.16.1 Perform CHANNEL CHECK.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.3.16.2 Perform CHANNEL FUNCTIONAL TEST.	Once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment
SR 3.3.16.3 Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.17 Emergency Power Switching Logic (EPSL) Automatic Transfer Function

LCO 3.3.17 Two channels of the EPSL Automatic Transfer Function shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Restore channel to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	12 hours 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.17.1 Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.18 Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits

LCO 3.3.18 Three channels of each of the following EPSL voltage sensing circuits shall be OPERABLE:

a. Startup Transformer;
b. Standby Bus 1;
c. Standby Bus 2; and
d. Auxiliary Transformer.

---------------------------NOTE---------------------------

1. If both N breakers are open, Auxiliary Transformer voltage sensing circuits are not required to be OPERABLE.

2. When not in MODES 1, 2, 3 and 4, only EPSL voltage sensing circuit(s) associated with required AC power source(s) are required to be OPERABLE.

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

ACTIONS

-----------------------------------NOTE-----------------------------------

Separate Condition entry is allowed for each circuit.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required circuits with one channel inoperable.	A.1 Restore channel to OPERABLE status.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met in MODES 1, 2, 3, and 4.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours 84 hours
C. Two or more channels of a required circuit inoperable when not in MODES 1, 2, 3, and 4. OR Required Action and associated Completion Time not met when not in MODES 1, 2, 3, and 4.	C.1 Declare affected AC power source(s) inoperable.	Immediately
D. Required Action and associated Completion Time not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.18.1 Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.19 Emergency Power Switching Logic (EPSL) 230 kV Switchyard Degraded Grid Voltage Protection (DGVP)

LCO 3.3.19 Three DGVP voltage sensing channels and two DGVP actuation logic channels shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One voltage sensing channel inoperable.	A.1 Place channel in trip.	72 hours
B. One actuation logic channel inoperable.	B.1 Restore channel 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. <u>AND</u> C.2 Be in MODE 5.	12 hours 84 hours
D. Two or more voltage sensing channels inoperable. <u>OR</u> Two actuation logic channels inoperable.	D.1 Declare the overhead emergency power path inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.19.1	Perform a CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.19.2	Perform a CHANNEL CALIBRATION of the voltage sensing channel with the setpoint allowable value as follows: Degraded voltage ≥ 219 kV and ≤ 222 kV with a time delay of 9 seconds ±1 second.	18 months

3.3 INSTRUMENTATION

3.3.20 Emergency Power Switching Logic (EPSL) CT-5 Degraded Grid Voltage Protection (DGVP)

LCO 3.3.20 Three CT-5 DGVP voltage sensing channels and two CT-5 DGVP actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4 when the Central Switchyard is energizing the standby buses.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One voltage sensing channel inoperable.	A.1 Place channel in trip.	72 hours
B. One actuation logic channel inoperable.	B.1 Restore channel to OPERABLE status.	72 hours
C. Two or more voltage sensing channels inoperable. <u>OR</u> Two actuation logic channels inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A or B not met.	C.1 Open SL breakers.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.20.1	Perform a CHANNEL FUNCTIONAL TEST.	18 months
SR 3.3.20.2	Perform a CHANNEL CALIBRATION of the voltage sensing channel with the setpoint allowable value as follows: a. Degraded voltage ≥ 4143 V and ≤ 4185 V with a time delay of 9 seconds ±1 second for the first level undervoltage inputs; and b. Degraded voltage ≥ 3871 V and ≤ 3901 V for the second level undervoltage inputs.	18 months

3.3 INSTRUMENTATION

3.3.21 Emergency Power Switching Logic (EPSL) Keowee Emergency Start Function

LCO 3.3.21 Two channels of the EPSL Keowee Emergency Start Function shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	12 hours 84 hours
C. Two channels inoperable.	C.1 Declare both Keowee Hydro Units inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.21.1 Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.22 Emergency Power Switching Logic (EPSL) Manual Keowee Emergency Start Function

LCO 3.3.22 One channel of the EPSL Manual Keowee Emergency Start Function shall be OPERABLE.

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	A.1 Declare both Keowee Hydro Units inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.22.1 Perform CHANNEL FUNCTIONAL TEST.	12 months

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.3 - INSTRUMENTATION

ATTACHMENT 2

BASES

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protective System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during anticipated transients. By tripping the reactor, the RPS also assists the Engineered Safeguards (ES) Systems in mitigating accidents.

The protective and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establishes the threshold for protective system action to prevent exceeding acceptable limits during accidents or transients.

During anticipated transients, which are those events expected to occur one or more times during the unit's life, the acceptable limit is:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value;

b. Fuel centerline melt shall not occur; and

c. The RCS pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 20 and 10 CFR 100 criteria during anticipated transients.

Accidents are events that are analyzed even though they are not expected to occur during the unit's life. The acceptable limit during accidents is that the offsite dose shall be maintained within reference 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

(continued)

BASES

BACKGROUND (continued)

RPS Overview

The RPS consists of four separate redundant protective channels that receive inputs of neutron flux, RCS pressure, RCS flow, RCS temperature, RCS pump status, reactor building (RB) pressure, main feedwater (MFW) pump status, and turbine status.

Figure 7.1, UFSAR, Chapter 7 (Ref. 1), shows the arrangement of a typical RPS protective channel. A protective channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and control rod drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protective System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protective System (RPS) – Reactor Trip Module (RTM)," and LCO 3.3.4, "control rod Drive (CRD) Trip Devices," discuss the remaining RPS elements.

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of any two trip signals in any of the four RPS channels will result in the trip of the reactor.

The Reactor Trip System (RTS) contains multiple CRD trip devices; two AC trip breakers, two DC trip breaker pairs, and eight electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker in series with a pair of DC breakers and functionally in series with four ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs. Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip.

The RPS consists of four independent protective channels, each containing an RTM. The RTM receives signals from its own measurement channels that indicate a protective channel

(continued)

BASES

BACKGROUND

RPS Overview (continued)

trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device.

The reactor is tripped by opening circuit breakers and ETA relays that interrupt the control power supply to the CRDs. Six breakers are installed to increase reliability and allow testing of the trip system. A one-out-of-two taken twice logic is used to interrupt power to the rods.

The RPS has three bypasses: a shutdown bypass, a dummy bistable and an RPS channel bypass. Shutdown bypass allows the withdrawal of safety rods for SDM availability and rapid negative reactivity insertion during unit cooldowns or heatups. The dummy bistable is used to bypass one or more functions (bistable trips) associated with one RPS Channel. The RPS Channel bypass allows one entire RPS channel to be taken out of service for maintenance and testing. Test circuits in the trip strings allow complete testing of all RPS trip Functions.

The RPS operates from the instrumentation channels discussed next. The specific relationship between measurement channels and protective channels differs from parameter to parameter. Three basic configurations are used:

a. Four completely redundant measurements (e.g., reactor coolant flow) with one channel input to each protective channel;

b. Four channels that provide similar, but not identical, measurements (e.g., power range nuclear instrumentation where each RPS channel monitors a different quadrant), with one channel input to each protective channel; and

c. Redundant measurements with combinational trip logic outside of the protective channels and the combined output provided to each protective channel (e.g., main feedwater pump trip instrumentation).

(continued)

BASES

BACKGROUND

RPS Overview (continued)

These arrangements and the relationship of instrumentation channels to trip Functions are discussed next to assist in understanding the overall effect of instrumentation channel failure.

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following trip Functions:

1. Nuclear Overpower

   a. Nuclear Overpower – High Setpoint;

   b. Nuclear Overpower – Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower Flux/Flow Imbalance;

9. Main Turbine Trip (Hydraulic Fluid Pressure); and

10. Loss of Main Feedwater (LOMFW) Pumps (Hydraulic Oil Pressure).

The power range instrumentation has four linear level channels, one for each core quadrant. Each channel feeds one RPS protective channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The difference of the top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE for the associated core quadrant.

(continued)

BASES

BACKGROUND (continued)

Reactor Coolant System Outlet Temperature

The Reactor Coolant System Outlet Temperature provides input to the following Functions:

2. RCS High Outlet Temperature; and

5. RCS Variable Low Pressure.

The RCS Outlet Temperature is measured by two resistance elements in each hot leg, for a total of four. One temperature detector is associated with each protective channel.

Reactor Coolant System Pressure

The Reactor Coolant System Pressure provides input to the following Functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

The RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each hot leg, for a total of four. One sensor is associated with each protective channel.

Reactor Building Pressure

The Reactor Building Pressure measurements provide input only to the Reactor Building High Pressure trip, Function 6. There are four RB High Pressure sensors, one associated with each protective channel.

(continued)

BASES

BACKGROUND (continued)

Reactor Coolant Pump Power Monitoring

Reactor coolant pump power monitors are inputs to the Reactor Coolant Pump to Power trip, Function 7. Each RCP, operating current, and voltage is measured by four current transformers and four potential transformers driving four underpower relays. Each power monitoring channel consists of an underpower relay. One channel for each pump is associated with each protective channel.

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower Flux/Flow Imbalance trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight high accuracy differential pressure transmitters, four on each loop, which measure flow through calibrated flow tubes. One flow input in each loop is associated with each protective channel.

Main Turbine Automatic Stop Oil Pressure

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip (Hydraulic Fluid Pressure) reactor trip, Function 9. Each of the four protective channels receives turbine status information from one of the four pressure switches monitoring main turbine automatic stop oil pressure. An open indication will be provided to the RPS on a turbine trip. Contact buffers in each protective channel continuously monitor the status of the contact inputs and initiate an RPS trip when a main turbine trip is indicated.

Feedwater Pump Hydraulic Oil Pressure

Feedwater Pump Hydraulic Oil Pressure is an input to the Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip, Function 10. Hydraulic Oil pressure is measured by four switches on each feedwater pump. One switch on each pump, connected in series with a switch on the other MFW pump, is associated with each protective channel.

(continued)

BASES

BACKGROUND (continued)

RPS Bypasses

The RPS is designed with three types of bypasses: dummy bistable, channel bypass and shutdown bypass.

The dummy bistable provides a method of placing one or more functions in a RPS protective channel in a bypassed condition, the channel bypass provides a method of placing all Functions in one RPS protective channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed next.

Dummy Bistable

The dummy bistable is used to bypass one or more functions (bistable trips) associated with one RPS Channel. A dummy bistable is used if a parameter in an RPS channel fails and causes that channel to trip. Dummy bistables may be used in only one RPS channel at a time. Also, if an RPS channel is bypassed, no other RPS channel may contain a dummy bistable. Inserting a dummy bistable in the place of a failed (tripped) bistable allows the RPS channels to be reset, thus allowing the remainder of the functions in that RPS channel to be returned to service. This is more conservative than manually bypassing the entire RPS channel. The trip functions in an RPS channel with a dummy bistable are not considered OPERABLE.

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protective channel trip relay energized regardless of the status of the instrumentation channel of the bistable relay contacts. To place a protective channel in channel bypass, the other channels must not be in channel bypass or otherwise inoperable. This can be verified by observing alarms/indicator lights. This is administratively controlled by having only one manual bypass key available for each unit. All RPS trips are reduced to a two-out-of-three logic in channel bypass.

(continued)

BASES

BACKGROUND (continued)

Shutdown Bypass

During unit cooldown and heatup, it is desirable to leave the safety rods at least partially withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

However, the unit is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at 1800 psig and the rods will fall into the core. To avoid this, the protective system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1800 psig. The RCS pressure is decreased to less than 1720 psig, then each RPS channel is placed in shutdown bypass.

In shutdown bypass, a normally closed contact opens when the operator closes the shutdown bypass key switch. This action bypasses the RCS Low Pressure trip, Nuclear Overpower Flux/Flow Imbalance trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, and inserts a new RCS High Pressure, 1720 psig trip. The operator can now withdraw the safety rods for additional rapidly insertable negative reactivity.

The insertion of the new high pressure trip performs two functions. First, with a trip setpoint of 1720 psig, the bistable prevents operation at normal system pressure, 2155 psig, with a portion of the RPS bypassed. The second function is to ensure that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. The safety rods are then withdrawn and remain at the full out condition for the rest of the heatup.

(continued)

BASES

BACKGROUND

Shutdown Bypass (continued)

In addition to the Shutdown Bypass RCS High Pressure trip, the high flux trip setpoint is administratively reduced to ≤ 5% RTP prior to placing the RPS in shutdown bypass. This provides a backup to the Shutdown Bypass RCS High Pressure trip and allows low power physics testing while preventing the generation of any significant amount of power.

Module Interlock and Test Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode, it causes the test trip relay to open and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested. Each trip module is electrically interlocked to the other three trip modules. Removal of a trip module will indicate a tripped channel in the remaining trip modules.

Trip Setpoints/Allowable Value

The trip setpoints are the normal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The trip setpoints used in the bistables are based on the analytical limits stated in UFSAR, Chapter 15 (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 3), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in Reference 4. The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL

(continued)

BASES

BACKGROUND (continued)

Trip Setpoints/Allowable Value (continued)

TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during anticipated transients and that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the anticipated transient or accident and the equipment functions as designed. Note that in LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 for Functions 1 through 8 and 11 are the LSSS.

Each channel can be tested online to verify that the setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. Surveillances for the channels are specified in the SR section.

The Allowable Values listed in Table 3.3.1-1 are based on the methodology described in Reference 4, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of those uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in the UFSAR, Chapter 15 (Ref. 2), takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high RCS temperature, turbine trip, and loss of main feedwater. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

performance. These Functions also serve as backups to Functions that were credited in the safety analysis.

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The three channels of each Function in Table 3.3.1-1 of the RPS instrumentation shall be OPERABLE during its specified Applicability to ensure that a reactor trip will be actuated if needed. Additionally, during shutdown bypass with any CRD trip breaker closed, the applicable RPS Functions must also be available. This ensures the capability to trip the withdrawn CONTROL RODS exists at all times that rod motion is possible. The trip Function channels specified in Table 3.3.1-1 are considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the setpoint calculations. Each allowable Value specified is more conservative than instrument uncertainties appropriate to the trip Function. These uncertainties are defined in Reference 4.

For most RPS Functions, the trip setpoint Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific values for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the minimum deviation from normal values that will indicate the

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

a. Nuclear Overpower - High Setpoint

The Nuclear Overpower - High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core neutron leakage flux.

The Nuclear Overpower - High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to prevent exceeding acceptable fuel damage limits.

Thus, the Nuclear Overpower - High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower Flux/Flow Imbalance, provide more direct protection. The role of the Nuclear Overpower - High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower - High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident and the rod ejection accident. By providing a trip during these events, the Nuclear Overpower - High Setpoint trip protects the unit

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Nuclear Overpower - High Setpoint (continued)

from excessive power levels and also serves to limit reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

b. Nuclear Overpower - Low Setpoint

Prior to initiating shutdown bypass, the Nuclear Overpower - Low Setpoint trip must be reduced to ≤ 5% RTP. The low power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the unit is protected from excessive power conditions when other RPS trips are bypassed.

The setpoint Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. RCS High Outlet Temperature

The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for DNB. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable Low Pressure trip. The RCS High Outlet Temperature trip provides steady state protection for the DNBR SL.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. RCS High Outlet Temperature (continued)

The RCS High Outlet Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip is not required to mitigate accidents that create harsh conditions in the RB.

3. RCS High Pressure

The RCS High Pressure trip works in conjunction with the pressurizer and main steam relief valves to prevent RCS overpressurization, thereby protecting the RCS High Pressure SL.

The RCS High Pressure trip has been credited in the transient analysis calculations for slow positive reactivity insertion transients (rod withdrawal transients and moderator dilution). The rod withdrawal transient covers a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower–High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The setpoint Allowable Value is selected to ensure that the RCS High Pressure SL is not challenged during steady state operation or slow power increasing transients. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs) and main steam line break (MSLB) accidents. Harsh RB conditions created by small break LOCAs cannot affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination.

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the RCS High Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is assumed for transient protection in the unit

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

5. RCS Variable Low Pressure (continued)

safety analysis but does not affect the limiting cases; therefore, determination of the setpoint Allowable Value does not account for errors induced by a harsh RB environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients inside containment. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of more than two RCPs.

(continued)

BASES

APPLICABLE SAFETY ANALYSIS, LCO, and APPLICABILITY

7. Reactor Coolant Pump to Power (continued)

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least three RCPs are operating. RCP status is monitored by power transducers on each pump. These relays indicate a loss of an RCP on underpower. The underpower setpoint is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power setpoint account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. Nuclear Overpower Flux/Flow Imbalance

The Nuclear Overpower Flux/Flow Imbalance trip provides steady state protection for the power imbalance SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the DNBR SL for the loss of one or more RCPs and for locked RCP rotor accidents.

The power to flow ratio of the Nuclear Overpower Flux/Flow Imbalance trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system flow rate is less than full four pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs when the core power, axial power peaking, and

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Nuclear Overpower Flux/Flow Imbalance (continued)

reactor coolant flow conditions indicate an approach to DNB or fuel centerline temperature limits. By measuring reactor coolant flow and by tripping only when conditions approach an SL, the unit can operate with the loss of one pump from a four pump initial condition at power levels at least as low as approximately 80% RTP. The Allowable Value for the Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

9. Main Turbine Trip (Hydraulic Fluid Pressure)

The Main Turbine Trip Function trips the reactor when the main turbine is lost at high power levels. The Main Turbine Trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine Trip Function was added to the B&W designed units in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS power operated relief valve (PORV) actuation for turbine trip cases. This trip is activated at higher power levels, thereby limiting the range through which the Integrated Control System must provide an automatic runback on a turbine trip.

Each of the four turbine hydraulic fluid pressure switches feeds one protective channel through buffers that continuously monitor the status of the contacts.

For the Main Turbine Trip (Hydraulic Fluid Pressure) bistable, the Allowable Value of 800 psig is selected to provide a trip whenever main turbine hydraulic fluid pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 30% RTP. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Loss of Main Feedwater Pumps (Hydraulic Oil Pressure)

The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip provides a reactor trip at high power levels when both MFW pumps are lost. The trip provides an early reactor trip in anticipation of the loss of heat sink associated with the LOMF. This trip was added in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip at high power levels for a LOMF to minimize challenges to the PORV.

For the feedwater pump hydraulic oil pressure bistables, the Allowable Value of 75 psig is selected to provide a trip whenever feedwater pump hydraulic oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 2% RTP. The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass RCS High Pressure is provided to allow for withdrawing the CONTROL RODS prior to reaching the normal RCS Low Pressure trip setpoint. The shutdown bypass provides trip protection during deboration and RCS heatup by allowing the operator to at least partially withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Use of the shutdown bypass trip requires that the neutron power trip setpoint be reduced to 5% of full power or less. The Shutdown Bypass RCS High Pressure trip forces a reactor trip to occur whenever the unit switches from power operation to shutdown bypass or vice versa. This ensures that the CONTROL RODS are all inserted before power operation can begin. The operator is required to remove the shutdown bypass, reset the Nuclear Overpower–High

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

Power trip setpoint, and again withdraw the safety group rods before proceeding with startup.

Accidents analyzed in the UFSAR, Chapter 15 (Ref. 2), do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the UFSAR.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of $\leq$ 1720 psig and the Nuclear Overpower – Low Setpoint set at or below 5% RTP, the trips listed below can be bypassed. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low Setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

1.a Nuclear Overpower – High Setpoint;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

The Shutdown Bypass RCS High Pressure Function's Allowable Value is selected to ensure a trip occurs before producing THERMAL POWER.

General Discussion

The RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 8).

In MODES 1 and 2, the following trips shall be OPERABLE because the reactor can be critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during anticipated transients and to

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

General Discussion (continued)

assist the ESPS in providing acceptable consequences during accidents.

1a. Nuclear Overpower – High Setpoint;

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance.

Functions 1, 3, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below 1720 psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower – Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

In MODE 3 when not operating in shutdown bypass but with any CRD trip breaker in the closed position and the CRD system capable of rod withdrawal, the Nuclear Overpower – High Setpoint trip and the RCS High Pressure trip are required to be OPERABLE.

Two Functions are required to be only OPERABLE during portions of MODE 1. These are the Main Turbine Trip (Hydraulic Fluid Pressure) and the Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) trip. These Functions are required to be OPERABLE at ≥ 30% RTP and ≥ 2% RTP, respectively. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

General Discussion (continued)

Because the safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5 if either the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

However, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower – Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower – Low setpoint trips are sufficient to prevent an approach to conditions that could challenge SLs.

ACTIONS

Conditions A and B are applicable to all RPS protective Functions. If a channel's trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and Condition A entered immediately.

A.1

For Required Action A.1, if one or more Functions in a required protective channel becomes inoperable, the affected protective channel must be placed in trip. This Required Action places all RPS Functions in a one-out-of-two logic configuration. The "non-required" channel is placed in bypass when the required inoperable channel is placed in trip to prevent bypass of a second required channel. In this configuration, the RPS can still perform its safety functions in the presence of a random failure of any single Channel. The 1 hour Completion Time is sufficient time to perform Required Action A.1.

(continued)

BASES

ACTIONS (continued)

B.1

Required Action B.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and the associated Completion Time of Condition A are not met or if more than two channels are inoperable, Condition B is entered to provide for transfer to the appropriate subsequent Condition.

C.1 and C.2

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition C, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and to open all CRD trip breakers without challenging unit systems.

D.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging unit systems.

E.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be

(continued)

BASES

ACTIONS

E.1 (continued)

OPERABLE. To achieve this status, THERMAL POWER must be reduced < 30% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 30% RTP from full power conditions in an orderly manner without challenging unit systems.

F.1

If the Required Action and associated Completion Time of Condition A are not met and Table 3.3.1-1 directs entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < 2% RTP. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach 2% RTP from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION testing.

The SRs are modified by a Note. The Note directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

The Frequency, equivalent to once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower Flux/Flow Imbalance Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 24 hours when reactor power is > 15% RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

channels are normalized to the calorimetric. If the calorimetric exceeds the Nuclear Instrumentation System (NIS) channel output by ≥ 2% RTP, the NIS is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. A Note clarifies that this Surveillance is required to be performed only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are less accurate.

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric exceeds the power range channel's output by ≥ 2% RTP. The value of 2% is adequate because this value is assumed in the safety analyses of UFSAR, Chapter 15 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds a small fraction of 2% in any 24 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is ≥ 15% RTP. A Note clarifies that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. If the absolute difference between the power range and incore measurements is ≥ 2% RTP, the power range channel is not inoperable, but a calibration that adjusts the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.3 (continued)

setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SR 3.3.1.4

This SR calibrates the power range channel output to the calorimetric coincident with the imbalance output being calibrated to the imbalance condition determined by the incore neutron detector system. A Note clarifies that this Surveillance is required to be performed only if reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are less accurate.

The 31 day Frequency specified for the Nuclear Overpower trip string considers drift determined using the methodology of Reference 4. Furthermore, operating experience shows the reliability of the trip string is acceptable when calibrated on this interval.

SR 3.3.1.5

A CHANNEL FUNCTIONAL TEST is performed on each required RPS channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The requirements for this review are outlined in BAW-10167 (Ref. 7).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.5 (continued)

The Frequency of 45 days on a STAGGERED TEST BASIS is consistent with the calculations of Reference 7 that indicate the RPS retains a high level of reliability for this test interval.

SR 3.3.1.6

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure virtually instantaneous response.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapter 7.

2. UFSAR, Chapter 15.

3. 10 CFR 50.49.

(continued)

BASES

REFERENCES (continued)

4. EDM-102, "Instrument Setpoint/Uncertainty Calculations."

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1979.

6. BAW-1893, "Basis for Raising Arming Threshold for Anticipating Reactor Trip on Turbine Trip," October 1985.

7. BAW-10167, May 1986.

8. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protective System (RPS) Manual Reactor Trip

BASES

BACKGROUND

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room. Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switch contacts, one for each train. This trip is independent of the automatic trip system. As shown in Figure 7.1, UFSAR, Chapter 7 (Ref. 1), power for the control rod drive (CRD) breaker undervoltage coils and contactor coils comes from the reactor trip modules (RTMs). The manual trip switch contacts are located between the RTM output and the breaker undervoltage coils. Opening of the switch contacts opens the lines to the breakers, tripping them. The switch contacts also energize the breaker shunt trip mechanisms. There is a separate switch contact in series, with the output of each of the four RTMs. All switch contacts are actuated through a mechanical linkage from a single push button.

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip Function is required as a backup to the automatic trip functions and allows operators to shut down the reactor.

The Manual Reactor Trip Function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO on the RPS Manual Reactor Trip requires that the trip shall be OPERABLE whenever the reactor is critical or any time any control rod breaker is closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any event that in the operator's judgment requires protective action, even if no automatic trip condition exists.

(continued)

BASES

LCO (continued)

The Manual Reactor Trip Function is composed of four electrically independent trip switch contacts sharing a common mechanical push button.

APPLICABILITY

The Manual Reactor Trip Function is required to be OPERABLE in MODES 1 and 2. It is also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breaker is in the closed position and if the CRD System is capable of rod withdrawal. The safety function of the RPS is to trip the CONTROL RODS; therefore, the Manual Reactor Trip Function is not needed in MODE 3, 4, or 5 if either the reactor trip breakers are open or if the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

ACTIONS

A.1

Condition A applies when the Manual Reactor Trip Function is found inoperable. One hour is allowed to restore Function to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour Completion Time is sufficient time to correct minor problems.

B.1 and B.2

With the Required Action and associated Completion Time not met in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

C.1

With the Required Action and associated Completion Time not met in MODE 4 or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that these Surveillances are only performed during a unit outage.

REFERENCES

1. UFSAR, Chapter 7.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protective System (RPS) – Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure 7.1, UFSAR, Chapter 7 (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and CONTROL ROD drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS – RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip devices.

The RPS trip scheme consists of series contacts that are operated by bistables. During normal unit operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When an RTM channel trip relay de-energizes, several things occur:

a. Each of the four (4) output logic relays "informs" its associated RPS channel that a reactor trip signal has occurred in the tripped RPS channel;

b. The contacts in the trip device circuitry, powered by the tripped channel, open, but the trip device remains energized through the closed contacts from the other RTMs. (This condition exists in each RPS – RTM. Each RPS – RTM controls power to a trip device.); and

c. The contact in parallel with the channel reset switch opens and the trip is sealed in. To re-energize the channel trip relay, the channel reset switch must be depressed after the trip condition has cleared.

(continued)

BASES

BACKGROUND (continued)

When the second RPS channel senses a reactor trip condition, the output logic relays for the second channel de-energize and open contacts that supply power to the trip devices. With contacts opened by two separate RPS channels, power to the trip devices is interrupted and the CONTROL RODS fall into the core.

A minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, because the bistable relay contacts for each function are in series with the channel trip relays, two channel trips caused by different trip functions can result in a reactor trip.

APPLICABLE SAFETY ANALYSES

Transient and accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip Functions in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation."

The RTMs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable.

An OPERABLE RTM must be able to receive and interpret trip signals from its own and other OPERABLE RPS channels and to open its associated trip device.

The requirement of four RTMs to be OPERABLE ensures that a minimum of two RTMs will remain OPERABLE if a single failure has occurred in one RTM and if a second RTM is out of service. This two-out-of-four trip logic also ensures that a single RTM failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

(continued)

BASES (continued)

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2. They are also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breakers are in the closed position and the CRD System is capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur, if needed. This condition can exist in all of these MODES; therefore, the RTMs must be OPERABLE.

ACTIONS

A.1.1, A.1.2, and A.2

When an RTM is inoperable, the associated CRD trip breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required Action A.1.1 or Required Action A.1.2 requires this either by tripping the CRD trip breaker or by removing power to the CRD trip device. Tripping one RTM or removing power opens one set of CRD trip devices. Power to hold up CONTROL RODS is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips.

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies if two or more RTMs are inoperable or if the Required Action and associated Completion Time of Condition A are not met in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with power from all CRD trip breakers removed within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Condition C applies if two or more RTMs are inoperable or if the Required Action and associated Completion Time of Condition A are not met in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing power from all CRD trip breakers. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove power from all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The SRs include performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals.

The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event.

Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each week. Testing one RTM each week reduces the likelihood of the same systematic test errors being introduced into each redundant RTM.

REFERENCES

1. UFSAR, Chapter 7.

2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.4 Control Rod Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protective System (RPS) contains multiple CRD trip devices: two AC trip breakers, two DC trip breaker pairs, and eight electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker in series with a pair of DC breakers and functionally in series with four ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD System.

Figure 7.1, UFSAR, Chapter 7 (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

Power to CRDs is supplied from two separate sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage trip coils are powered by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC power supplies and the regulating rod, APSR and auxiliary power supplies.

The DC power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A; the other, phase CC. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls half of the power to two of the four safety rod groups. The undervoltage trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C. The other two breakers are controlled by RPS channel D.

In addition to the DC power supplies, the redundant buses also supply power to the regulating rod, APSR and auxiliary power supplies. These power supplies contain silicon controlled rectifiers (SCRs) that are gated on and off to

(continued)

BASES

BACKGROUND (continued)

provide power to, and remove power from, the phases of the CRD mechanisms. The gating control signal for these SCRs is supplied through the closed contacts of the ETA relays. These contacts are referred to as E and F contactors, and are controlled by the C and D RPS channels respectively.

The AC breaker and DC breakers are in series in one of the power supplies; whereas, the redundant AC breaker and DC breakers are in series in the other power supply to the CONTROL RODS. The logic required to cause a reactor trip is the opening of a circuit breaker in each of the redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

a. If the A AC circuit breaker opens:

   1. the input power to associated DC power supply is lost, and

   2. the SCR supply from the associated power source is lost.

b. If the D DC circuit breaker(s) and F contactors open:

   1. the output of the DC power supply is lost, and

   2. when the F contactor opens, SCR gating power is lost.

c. The combination of (a) and (b) causes a reactor trip.

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable lower pressure bistable relay de-energizes the

(continued)

BASES

BACKGROUND (continued)

channel C trip relay. When the output logic relays in channel B and C de-energize, the B and C contacts in the trip logic of each channel's reactor trip module (RTM) open causing an undervoltage to each trip breaker. All trip breakers and the ETA relay contactors open, and power is removed from all CRD mechanisms. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The CONTROL ROD position limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protective System (RPS) Instrumentation."

The CRD trip devices satisfy Criterion 3 of CFR 50.36 (Ref. 2).

LCO

The LCO requires all of the specified CRD trip devices to be OPERABLE. Failure of any required CRD trip device renders a portion of the RPS inoperable and reduces the reliability of the affected Functions. Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip may not reliably occur when initiated either automatically or manually.

All required CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting power to the CRDs. Both of the CRD trip breaker's diverse trip devices and the breaker itself must be functioning properly for the breaker to be OPERABLE.

(continued)

BASES

LCO (continued)

Both ETA relays associated with each of the three regulating rod groups and the two ETA relays associated with the auxiliary power supply must be OPERABLE to satisfy the LCO. The ETA relays associated with the APSR power supply are not required to be OPERABLE because the APSRs are not designed to fall into the core upon initiation of a reactor trip.

Requiring all breakers and ETA relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure.

APPLICABILITY

The CRD trip devices shall be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

The CRD trip devices are designed to ensure that a reactor trip would occur if needed. Since this condition can exist in all of these MODES, the CRD trip devices shall be OPERABLE.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each CRD trip device.

A.1 and A.2

Condition A represents reduced redundancy in the CRD trip Function. Condition A applies when:

- One diverse trip Function (undervoltage or shunt trip device) is inoperable in one or more CRD trip breaker(s) or breaker pair; or
- One diverse trip Function is inoperable in both DC trip breakers associated with one protective channel. In this case, the inoperable trip Function does not need to be the same for both breakers.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

If one of the diverse trip Functions on a CRD trip breaker or breaker pair becomes inoperable, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually tripping the inoperable CRD trip breaker or by removing power from the inoperable CRD trip breaker. Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single failure from preventing a reactor trip. The 48 hour Completion Time has been shown to be acceptable through operating experience.

B.1 and B.2

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when both diverse trip Functions are inoperable in one or more trip breaker(s) or breaker pairs.

Required Action B.1 and Required Action B.2 are the same as Required Action A.1 and Required Action A.2, but the Completion Time is shortened. The 1 hour Completion Time allowed to trip or remove power from the CRD trip breaker allows the operator to take all the appropriate actions for the inoperable breaker and still ensures that the risk involved is acceptable.

C.1 and C.2

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of two actions to eliminate reliance on the failed ETA relay. This first option is to switch the affected CONTROL ROD group to an alternate power supply. This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. The second option is to trip the corresponding AC CRD trip breaker. This results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence. The 1 hour Completion Time is sufficient to perform the Required Action.

(continued)

BASES

ACTIONS (continued)

D.1, D.2.1, and D.2.2

With the Required Action and associated Completion Time of Condition A, B, or C not met in MODE 1, 2, or 3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3, with all CRD trip breakers open or with power from all CRD trip breakers removed within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

E.1 and E.2

With the Required Action and associated Completion Time of Condition A, B, or C not met in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or power from all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove power from all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 31 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the trip breakers. The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapter 7.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safeguards Protective System (ESPS) Analog Instrumentation

BASES

BACKGROUND

The ESPS initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and to mitigate accidents.

ESPS actuates the following systems:

- High pressure injection (HPI);
- Low pressure injection (LPI);
- Reactor building (RB) cooling;
- Penetration room ventilation;
- RB Spray;
- RB Isolation; and
- Keowee Hydro Unit Emergency Start.

The ESPS operates in a distributed manner to initiate the appropriate systems. The ESPS does this by determining the need for actuation in each of three analog channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to digital automatic actuation logic channels, which perform the two-out-of-three logic to determine the actuation of each end device. Each end device has its own automatic actuation logic, although all digital automatic actuation logic channels take their signals from the same bistable in each channel for each Parameter.

Four Parameters are used for actuation:

- Low Reactor Coolant System (RCS) Pressure;
- Low Low RCS Pressure;
- High RB Pressure; and
- High High RB Pressure.

(continued)

BASES

BACKGROUND (continued)

LCO 3.3.5 covers only the analog instrumentation channels that measure these Parameters. These channels include all intervening equipment necessary to produce actuation before the measured process Parameter exceeds the limits assumed by the accident analysis. This includes sensors, bistable devices, operational bypass circuitry, and output relays. LCO 3.3.6, "Engineered Safeguards Protective System (ESPS) Manual Initiation," and LCO 3.3.7, "Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels," provide requirements on the manual initiation and digital automatic actuation logic Functions.

The ESPS contains three analog channels. Each analog channel provides input to digital logic channels that initiate equipment with a two-out-of-three logic on each digital logic channel. Each analog channel includes inputs from one analog instrumentation channel of Low RCS Pressure, Low Low RCS Pressure, High RB Pressure, and High High RB Pressure. Digital automatic actuation logic channels combine the three analog channel trips to actuate the individual Engineered Safeguards (ES) components needed to initiate each ES System. Figure 7.5, UFSAR, Chapter 7 (Ref. 1), illustrates how analog instrumentation channel trips combine to cause digital logic channel trips.

The following matrix identifies the analog instrumentation (measurement) channels and the Digital Automatic Actuation Logic Channels actuated by each.

Digital Logic Channels	Actuated Systems/ Functions	RCS PRESS LOW	RCS PRESS LOW LOW	RB PRESS HIGH	RB PRESS HIGH HIGH
1 and 2	HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	X		X	
3 and 4	LPI and RB Essential isolation		X	X	
5 and 6	RB Cooling, RB Essential isolation, and Penetration Room Vent.			X	
7 and 8	RB Spray				X

(continued)

BASES

BACKGROUND (continued)

The ES equipment is generally divided between the two redundant digital actuation logic channels. The division of the equipment between the two digital actuation logic channels is based on the equipment redundancy and function and is accomplished in such a manner that the failure of one of the digital actuation logic channels and the related safeguards equipment will not inhibit the overall ES Functions. Redundant ES pumps are controlled from separate and independent digital actuation logic channels with the exception of HPI B pump which is actuated by both.

The actuation of ES equipment is also available by manual actuation switches located on the control room console or ES panel.

The ESPS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate accidents, specifically the loss of coolant accident (LOCA) and main steam line break (MSLB) events. The ESPS relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems of LCO 3.3.7.

Engineered Safeguards Protective System Bypasses

No provisions are made for maintenance bypass of ESPS instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow unit shutdown without spurious ESPS actuation.

The ESPS RCS pressure instrumentation channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low and low low pressure trips are required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded. This bypass provides an operational provision only outside the Applicability for this parameter, and provides no safety function.

(continued)

BASES

BACKGROUND (continued)

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protective System (RPS). Each of the pressure signals generated by these transmitters is monitored by four bistables to provide two trip signals, at ≥ 1500 psig and ≥ 500 psig, and two bypass permissive signals, at ≥ 1750 psig and ≥ 900 psig.

The outputs of the three bistables, associated with the low RCS pressure, ≥ 1500 psig, trip drive relays in two sets of identical and independent channels. These two sets of HPI channels each use a two-out-of-three coincidence network for HPI Actuation. The outputs of the three bistables associated with the Low Low RCS Pressure 500 psig trip drive relays in two sets of identical and independent channels. These two sets of LPI channels each use a two-out-of-three coincidence networks for LPI Actuation. The outputs of the three Low Low RCS Pressure bistables also trip the drive relays in the corresponding HPI Actuation channel as previously described.

Reactor Building Pressure

There are three Reactor Building pressure sensors. The output of each sensor terminates in an input isolation amplifier, which provides individually isolated outputs. One isolated output of each pressure measurement goes to the unit computer for monitoring. One output of each pressure measurement goes to a bistable which initiates action when its high building pressure trip point is exceeded. Each input isolation amplifier module contains an analog meter for indicating the measured pressure. Each of the three bistables has contact outputs that are combined in series with the output of the High and Low Pressure Injection System bistables as previously described.

The outputs of the three bistables are brought together in two identical two-out-of-three coincidence logics which provide two ESPS channels. Either of the two channels is independently capable of initiating the required protective action.

(continued)

BASES

BACKGROUND

Reactor Building Pressure (continued)

The ESPS channels of the Reactor Building Spray System are formed by two identical two-out-of-three logic networks with the active elements originating in six Reactor Building pressure sensing pressure switches.

Three independent pressure switches containing normally open contacts from one protective channel's two-out-of-three logic inputs. Three other identical pressure switches from the two-out-of-three logic inputs of the second protective channel. Either of the two protective channels is capable of initiating the required protective action.

Trip Setpoints and Allowable Values

Trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The trip setpoints used in the bistables are selected such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment induced errors for those ESPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2), the Allowable Values specified in Table 3.3.5-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their uncertainties, is provided in the Reference 3. The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Setpoints, in accordance with the Allowable Values, ensure that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident and the equipment functions as designed.

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

Each channel can be tested online to verify that the setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal may be injected in place of the field instrument signal.

APPLICABLE SAFETY ANALYSES

The following ESPS Functions have been assumed within the accident analyses.

High Pressure Injection

The ESPS actuation of HPI has been assumed for core cooling in the LOCA analysis and is credited with boron addition in the MSLB analysis.

Low Pressure Injection

The ESPS actuation of LPI has been assumed for large break LOCAs.

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESPS actuation of the RB coolers and RB Spray have been credited in RB analysis for LOCAs, both for RB performance and equipment environmental qualification pressure and temperature envelope definition. Accident dose calculations have credited RB Isolation and RB Spray.

Penetration Room Ventilation Actuation

The ESPS actuation of the penetration room ventilation system has been assumed for LOCAs. Accident dose calculations have credited penetration room ventilation.

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BASES

APPLICABLE SAFETY ANALYSES (continued)

Keowee Hydro Unit Emergency Start

The ESPS initiated Keowee Hydro Unit Emergency Start has been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small and large break LOCA analyses assume a conservative 48 second delay time for the actuation of HPI and LPI in UFSAR, Chapter 15 (Ref. 4). This delay time includes allowances for Keowee Hydro Unit starting, Emergency Core Cooling Systems (ECCS) pump starts, and valve openings. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed.

Accident analyses rely on automatic ESPS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, and MSLB events that result in RCS inventory reduction or severe loss of RCS cooling.

The ESPS channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 5).

LCO

The LCO requires three analog channels of ESPS instrumentation for each Parameter in Table 3.3.5-1 to be OPERABLE in each ESPS digital automatic actuation logic channel. Failure of any instrument renders the affected analog channel(s) inoperable and reduces the reliability of the affected Functions.

Only the Allowable Value is specified for each ESPS Function in the LCO. Nominal trip setpoints are specified in the setpoint calculations. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties

(continued)

BASES

LCO (continued)	appropriate to the trip Parameter. These uncertainties are defined in Reference 3.

The Allowable Values for bypass removal functions are stated in the Applicable MODES or Other Specified Condition column of Table 3.3.5-1.

Three ESPS analog instrumentation channels shall be OPERABLE to ensure that a single failure in one analog channel will not result in loss of the ability to automatically actuate the required safety systems.

The bases for the LCO on ESPS Parameters include the following.

Three analog channels of RCS Pressure – Low, RCS Pressure – Low Low, RB Pressure – High and RB Pressure – High High are required OPERABLE. Each analog channel includes a sensor, trip bistable, bypass bistable, bypass relays, and output relays. Failure of a bypass bistable or bypass circuitry, such that an analog channel cannot be bypassed, does not render the analog channel inoperable since the analog channel is still capable of performing its safety function, i.e., this is not a safety related bypass function.

APPLICABILITY

Three analog channels of ESPS instrumentation for each of the following Parameters shall be OPERABLE.

1. <u>Reactor Coolant System Pressure – Low</u>

   The RCS Pressure – Low actuation Parameter shall be OPERABLE during operation at or above 1750 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 1750 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required.

   The allowance for the bypass is consistent with the transition of the unit to a lower energy state,

(continued)

BASES

APPLICABILITY

1. Reactor Coolant System Pressure - Low (continued)

providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. RCS pressure and temperature are very low, and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

2. Reactor Coolant System Pressure - Low Low

The RCS Pressure - Low Low actuation Parameter shall be OPERABLE during operation above 900 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 900 psig, the low low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety system actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

(continued)

BASES

APPLICABILITY

2. Reactor Coolant System Pressure - Low Low (continued)

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. RCS pressure and temperature are very low, and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

3, 4. Reactor Building Pressure - High and Reactor Building Pressure - High High

The RB Pressure - High and RB Pressure - High High actuation Functions of ESPS shall be OPERABLE in MODES 1, 2, 3, and 4 when the potential for a HELB exists. In MODES 5 and 6, the unit conditions are such that there is insufficient energy in the primary and secondary systems to raise the containment pressure to either the RB Pressure - High or RB Pressure - High High actuation setpoints. Furthermore, in MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. RCS pressure and temperature are very low and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

ACTIONS

Required Actions A and B apply to all ESPS analog instrumentation Parameters listed in Table 3.3.5-1.

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Parameter.

If an analog channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing

(continued)

BASES

ACTIONS (continued)

electronics, or ESPS bistable is found inoperable, then all affected functions provided by that analog channel should be declared inoperable and the unit must enter the Conditions for the particular protective Parameter affected.

A.1

Condition A applies when one analog channel becomes inoperable in one or more Parameters. If one ESPS analog instrument channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another analog channel were to fail, the ESPS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays are tripped. This can normally be accomplished by tripping the affected bistables.

The 1 hour Completion Time is sufficient time to perform the Required Action.

B.1, B.2.1, B.2.2, and B.2.3

Condition B applies when the Required Action and associated Completion Time of Condition A are not met or when one or more parameters have two or more inoperable analog channels. If Condition B applies, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and, for the RCS Pressure – Low Parameter, to < 1750 psig, for the RCS Pressure – Low Low Parameter, to < 900 psig, and for the RB Pressure – High Parameter and RB Pressure – High High Parameter, to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The ESPS Parameters listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION. The operational bypasses associated with each RCS Pressure ESPS instrumentation channel are also subject

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

to these SRs to ensure OPERABILITY of the ESPS instrumentation channel.

SR 3.3.5.1

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that analog instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two analog instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

The Frequency, equivalent to every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel operability during normal operational use of the displays associated with the LCO's required channels.

SR 3.3.5.2

A Note permits delaying entry into applicable Conditions and Required Actions for up to 8 hours while bypassed for Surveillance testing provided the remaining two ESPS analog instrument channels are OPERABLE or tripped. The Note allows channel bypass for testing without entering the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2 (continued)

Conditions and Required Actions, although during this time period it cannot initiate ESPS. This allowance is based on the inability to perform the Surveillance in the time permitted by the Required Actions. Eight hours is sufficient time to perform the Surveillance. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing.

A CHANNEL FUNCTIONAL TEST is performed on each required ESPS analog channel to ensure the entire channel, including the bypass function, will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.

The Frequency of 31 days is based on operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the analog instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION assures that measurement errors and bistable setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

This Frequency is justified by the assumption of an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapter 7.
2. 10 CFR 50.49.

(continued)

BASES

REFERENCES (continued)

3. EDM-102, "Instrument Setpoint/Uncertainty Calculations."
4. UFSAR, Chapter 15.
5. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered Safeguards Protective System (ESPS) Manual Initiation

BASES

BACKGROUND

The ESPS manual initiation capability allows the operator to actuate ESPS Functions from the main control room in the absence of any other initiation condition. This ESPS manual initiation capability is provided in the event the operator determines that an ESPS Function is needed and has not been automatically actuated. Furthermore, the ESPS manual initiation capability allows operators to rapidly initiate Engineered Safeguards (ES) Functions.

LCO 3.3.6 covers only the system level manual initiation of these Functions. LCO 3.3.5, "Engineered Safeguards Protective System (ESPS) Analog Instrumentation," and LCO 3.3.7, "Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels," provide requirements on the portions of the ESPS that automatically initiate the Functions described earlier.

The ESPS manual initiation Function relies on the OPERABILITY of the digital automatic actuation logic channels (LCO 3.3.7) to perform the actuation of the systems. A manual trip push button is provided on the control room console for each of the digital automatic actuation logic channels. Operation of the push button energizes relays whose contacts perform a logical "OR" function with the automatic actuation.

The ESPS manual initiation channel is defined as the instrumentation between the console switch and the digital automatic actuation logic channel, which actuates the end devices. Other means of manual initiation, such as controls for individual ES devices, may be available in the control room and other unit locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.

APPLICABLE SAFETY ANALYSES

The ESPS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate accidents, specifically, the loss of coolant accident and steam line break events.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ESPS manual initiation ensures that the control room operator can rapidly initiate ES Functions. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESPS whenever any parameter is rapidly trending toward its trip setpoint.

The ESPS manual initiation functions satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

Two ESPS manual initiation channels of each ESPS Function shall be OPERABLE whenever conditions exist that could require ES protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESPS Function. The ESPS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

The required Function is provided by two associated channels as indicated in the following table:

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling, RB Essential isolation, and Penetration Room Vent.	5 & 6
RB Spray	7 & 8

(continued)

BASES (continued)

APPLICABILITY

The ESPS manual initiation Functions shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated engineered safeguard equipment is required to be OPERABLE. The manual initiation channels are required because ES Functions are designed to provide protection in these MODES. ESPS initiates systems that are either reconfigured for decay heat removal operation or disabled while in MODES 5 and 6. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and to respond by manually operating the ES components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESPS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more ESPS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. The Completion Time of 72 hours is based on operating experience and administrative controls, which provide alternative means of ESPS Function initiation via individual component controls. The 72 hour Completion Time is generally consistent with the allowed outage time for the safety systems actuated by ESPS.

B.1 and B.2

With the Required Action and associated Completion Time not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the ESPS manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the automatic actuation logic channels. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.7 Engineered Safeguards Protective System (ESPS) Digital Automatic Actuation Logic Channels

BASES

BACKGROUND

The digital automatic actuation logic channels of ESPS are defined as the instrumentation from the buffers of the ESPS analog instrument channels through the unit controllers that actuate ESPS equipment. Each of the components actuated by the ESPS Functions is associated with one or more digital automatic actuation logic channels. If two-out-of-three ESPS analog instrumentation channels indicate a trip, or if channel level manual initiation occurs, the digital automatic actuation logic channel is activated and the associated equipment is actuated. The purpose of requiring OPERABILITY of the ESPS digital automatic actuation logic channels is to ensure that the Functions of the ESPS can be automatically initiated in the event of an accident. Automatic actuation of some Functions is necessary to prevent the unit from exceeding the Emergency Core Cooling Systems (ECCS) limits in 10 CFR 50.46 (Ref. 1). It should be noted that OPERABLE digital automatic actuation logic channels alone will not ensure that each Function can be activated; the analog instrumentation channels and actuated equipment associated with each Function must also be OPERABLE to ensure that the Functions can be automatically initiated during an accident.

LCO 3.3.7 covers only the digital automatic actuation logic channels that initiates these Functions. LCO 3.3.5, "Engineered Safeguards Protective System (ESPS) Analog Instrumentation," and LCO 3.3.6, "Engineered Safeguards Protective System (ESPS) Manual Initiation," provide requirements on the analog instrumentation and manual initiation channels that input to the digital automatic actuation logic channels.

The ESPS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate accidents, specifically, the loss of coolant accident (LOCA) and main steam line break (MSLB) events. The ESPS relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems.

(continued)

BASES

BACKGROUND (continued)

The small and large break LOCA analyses assume a conservative 48 second delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in UFSAR, Chapter 15 (Ref. 2). This delay time includes allowances for Keowee Hydro Unit startup and loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

The ESPS automatic initiation of Engineered Safeguards (ES) Functions to mitigate accident conditions is assumed in the accident analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, and RB Isolation.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESPS actuation for protection of the core and RB and for limiting off site dose levels following an accident. The digital automatic actuation logic is an integral part of the ESPS.

The ESPS digital automatic actuation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 3).

LCO

The digital automatic actuation logic channels are required to be OPERABLE whenever conditions exist that could require ES protection of the reactor or the RB. This ensures automatic initiation of the ES required to mitigate the consequences of accidents.

The required Function is provided by two associated digital channels as indicated in the following table:

(continued)

BASES

LCO
(continued)

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling, RB Essential isolation, and Penetration Room Vent.	5 & 6
RB Spray	7 & 8

APPLICABILITY

The digital automatic actuation logic channels shall be OPERABLE in MODES 1 and 2 and in MODES 3 and 4 when the associated engineered safeguard equipment is required to be OPERABLE, because ES Functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by the ESPS are either reconfigured for decay heat removal operation or disabled. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and respond by manually operating the ES components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESPS digital automatic actuation logic channel.

A.1 and A.2

When one or more digital automatic actuation logic channels are inoperable, the associated component(s) can be placed in their engineered safeguard configuration. Required Action A.1 is equivalent to the digital automatic actuation logic channel performing its safety function ahead of time.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

In some cases, placing the component in its engineered safeguard configuration would violate unit safety or operational considerations. In these cases, the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its engineered safeguard configuration include, but are not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, or isolation of fluid systems that are normally functioning. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

Required Action A.2 requires declaring the associated components of the affected supported systems inoperable, since the true effect of digital automatic actuation logic channel failure is inoperability of the supported system. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component. A combination of Required Actions A.1 and A.2 may be used for different components associated with an inoperable digital automatic actuation logic channel.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31 day Frequency. The test demonstrates that each digital automatic actuation logic channel successfully performs the two-out-of-three logic combinations every 31 days. The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the automatic actuation logic. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50.46.
2. UFSAR, Chapter 15.
3. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.8 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events.

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed, and so that the need for and magnitude of further actions can be determined. These essential instruments are identified by the ONS specific Regulatory Guide 1.97 analysis (Ref. 1), UFSAR, Section 7.5 (Ref. 2), and the NRC's Safety Evaluation Report for the ONS Regulatory Guide 1.97 analysis (Ref. 3) which address the recommendations of Regulatory Guide 1.97 (Ref. 4), as required by Supplement 1 to NUREG-0737 (Ref. 5).

The instrument channels required to be OPERABLE by this LCO equate to two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category 1 variables.

Type A variables are specified because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for accidents.

Category 1 variables are the key variables deemed risk significant because they are needed to:

- Determine whether systems important to safety are performing their intended functions;

(continued)

BASES

BACKGROUND (continued)

- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and
- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

These key variables are identified by the ONS specific Regulatory Guide 1.97 analysis (Ref. 1). This analysis identifies the unit specific Type A and Category 1 variables and provides justification for deviating from the NRC proposed list of Category 1 variables.

The specific instrument Functions listed in Table 3.3.8-1 are discussed in the LCO Bases Section.

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the availability of information so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of accidents (e.g., loss of coolant accident (LOCA));
- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, which are required for safety systems to accomplish their safety functions;
- Determine whether systems important to safety are performing their intended functions;
- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and estimate the magnitude of any impending threat.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ONS specific Regulatory Guide 1.97 analysis (Ref. 1) documents the process that identifies Type A and Category 1 non-Type A variables.

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of 10 CFR 50.36 (Ref. 6). Category 1, non-type A, instrumentation must be retained in Technical Specifications because it is intended to assist operators in minimizing the consequences of accidents. Category 1, non-Type A variables are important for reducing public risk, and therefore, satisfy Criterion 4 of 10 CFR 50.36 (Ref. 6).

LCO

LCO 3.3.8 requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the unit and to bring the unit to, and maintain it in, a safe condition following that accident. Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information.

Where a channel includes more than one control room indication, such as both an indicator and a recorder, the channel is OPERABLE when at least one indication is OPERABLE.

The exception to the two channel requirement is containment isolation valve position. In this case, the important information is the status of the containment penetrations. The LCO requires one position indicator for each electrically controlled containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the electrically controlled valve and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

Each of the specified instrument Functions listed in Table 3.3.8-1 are discussed below:

(continued)

BASES

LCO (continued)

1. Wide Range Neutron Flux

Wide Range Neutron Flux indication is a Type B, Category 1 variable provided to verify reactor shutdown. The Wide Range Neutron Flux channels consist of two channels of fission chamber based instrumentation with readout on one recorder. (Note: four channels are available only two are required). The channels provide indication over a range of 1E-8% to 200% RTP.

2. Reactor Coolant System (RCS) Hot Leg Temperature

RCS Hot Leg Temperature instrumentation is a Type B, Category 1 variable provided for verification of core cooling and long term surveillance. The two channels provide readout on two indicators. Control room display is through the inadequate core cooling monitoring system. The channels provide indication over a range of 50°F to 700°F.

3, 5. Reactor Vessel Head Level and RCS Hot Leg Level

Reactor Vessel Water Level instrumentation is a Type B, Category 1 variable provided for verification and long term surveillance of core cooling. The reactor vessel level monitoring system provides an indication of the liquid level from the top of the Hot Leg on each steam generator to the bottom of the Hot Leg as it exits the vessel and from the top of the reactor vessel head to the bottom of the Hot Leg as it exits the vessel. Compensation is provided for impulse line temperature variations.

The Reactor Vessel Water Level channels consist of two Reactor Vessel Head Level channels that provide readout on two indicators (RC-LT0125 and RC-LT0126) with one channel recorded in the control room and two RCS Hot Leg Level channels that provide readout on two indicators (RC-LT0123 and RC-LT0124) with one channel recorded in the control room.

(continued)

BASES

LCO (continued)

4. <u>RCS Pressure (Wide Range)</u>

RCS Pressure (Wide Range) instrumentation is a Type A, Category 1 variable provided for verification of core cooling and RCS integrity long term surveillance.

Wide range RCS loop pressure is measured by pressure transmitters with a span of 0 psig to 3000 psig. The pressure transmitters are located outside the RB. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the inadequate core cooling plasma display. The inadequate core cooling plasma display is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument string.

RCS Pressure is a Type A, Category 1 variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator (SG) tube rupture or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting SG pressure or level, would use this indication. In addition, high pressure injection (HPI) flow is throttled based on RCS Pressure and subcooled margin. For some small break LOCAs, low pressure injection (LPI) may actuate with RCS pressure stabilizing above the shutoff head of the LPI pumps. If this condition exists, the operator is instructed to verify HPI flow and then terminate LPI flow prior to exceeding 30 minutes of LPI pump operation against a deadhead pressure. RCS Pressure, in conjunction with LPI flow, is also used to determine if a core flood line break has occurred.

6. <u>Containment Sump Water Level (Wide Range)</u>

Containment Sump Water Level (Wide Range) instrumentation is a Type B, Category 1 variable provided for verification and long term surveillance of RCS integrity. The Containment Sump Water Level instrumentation consists of two channels with readout on two indicators (LT-90 and LT-91) and one recorder. The indicated range is 0 to 15 feet.

(continued)

BASES

LCO (continued)

7. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) instrumentation is a Type B, Category 1 variable provided for verification of RCS and containment OPERABILITY. Containment Pressure instrumentation consists of two channels with readout on two indicators (PT-230 and PT-231) and one channel recorded. The indicated range is - 5.0 psig to 175 psig.

8. Containment Isolation Valve Position

Containment isolation valve (CIV) position is a Type B, Category 1 variable provided for verification of electrically controlled containment isolation valve position. In the case of CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each electrically controlled CIV in a containment penetration flow path, i.e., two total channels of CIV position indication for a penetration flow path with two electrically controlled valves. For containment penetrations with only one electrically controlled CIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the electrically controlled valve, as applicable, and prior knowledge of passive valve or system boundary status. As indicated by Note (a) to the Required Channels, if a penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status. Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. Note (c) to the Required Channels indicates that position indication requirements apply only to CIVs that are electrically controlled. The CIV position PAM instrumentation consists of limit switches that operate both Closed-Not Closed and

(continued)

BASES

LCO

8. Containment Isolation Valve Position (continued)

Open-Not Open control switch indication via indicating lights in the control room.

9. Containment Area Radiation (High Range)

Containment Area Radiation (High Range) instrumentation is a Type C, Category 1 variable provided to monitor the potential for significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. The Containment Area Radiation instrumentation consists of two channels (RIA 57 and 58) with readout on two indicators and one channel recorded. The indicated range is 1 to 10^7 R/hr.

10. Containment Hydrogen Concentration

Containment Hydrogen Concentration instrumentation is a Type A, Category 1 variable provided to detect high hydrogen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. The Containment Hydrogen Concentration instrumentation consists of two channels with readout on two indicators and one channel recorded. The indicated range is 0 to 10% hydrogen concentration.

11. Pressurizer Level

Pressurizer Level instrumentation is a Type A, Category 1 variable used in combination with other system parameters to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition. The Pressurizer Level instrumentation consists of three channels (two for Train A and one for Train B) on the

(continued)

BASES

LCO

11. Pressurizer Level (continued)

computer and one channel recorded (selected among the three channels). The indicated range is 0 to 400 inches (11% to 84% level as a percentage of volume).

12. Steam Generator Water Level

Steam Generator Water Level instrumentation is a Type A, Category 1 variable provided to monitor operation of decay heat removal via the SG. The indication of SG level is the extended startup range level instrumentation, covering a span of 0 inches to 388 inches above the lower tubesheet.

The operator relies upon SG level information following an accident (e.g., main steam line break, steam generator tube rupture) to isolate the affected SG to confirm adequate heat sinks for transients and accidents.

The extended startup range Steam Generator Level instrumentation consists of four indicators (2 per steam generator). The channels also display on the computer and one channel provides input to a recorder.

13. Steam Generator Pressure

Steam Generator Pressure instrumentation is a Type A, Category 1 variable provided to support operator diagnosis of a main steam line break or SG tube rupture accident to identify and isolate the affected SG. In addition, SG pressure is a key parameter used by the operator to evaluate primary-to-secondary heat transfer.

Steam generator pressure measurement is provided by two pressure transmitters per SG. Each instrument channel inputs to the ICCM cabinet that provide safety inputs to two indicators located on the main control board in the control room. One channel per SG also provides input to a recorder located in the control room.

(continued)

BASES

LCO (continued)

14. Borated Water Storage Tank (BWST) Level

BWST Level instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements, i.e., to determine when to initiate the switch over of the core cooling pump suction from the BWST to sump recirculation. BWST level measurement is provided by two channels with readout on two indicators and one channel recorded. The channels provide level indication over a range of 0 to 50 feet (13% to 100% of volume).

15. Upper Surge Tank (UST) Level

Upper Surge Tank Level instrumentation is a Type A, Category 1 variable provided to ensure a water supply for EFW. EFW draws condensate grade suction from the USTs and the Condenser Hotwell.

Two Category 1 instrumentation channels are provided for monitoring UST level. These instrument channels are inputs to corresponding train A and B Inadequate Core Cooling Monitoring (ICCM) system cabinets. The ICCM Train A cabinet provides UST level input to a dedicated qualified recorder and to a qualified indicator, both located in the Control Room. The ICCM Train B cabinet also provides an input to a qualified indicator located in the Control Room. The range of UST level indication is 0 to 12 feet.

UST Level is the primary indication used by the operator to identify loss of UST volume. The operator can then decide to replenish the UST or align suction to the EFW pumps from the hotwell.

16. Core Exit Temperature

Core Exit Temperature is a Type A, Category 1 variable provided for verification and long term surveillance of core cooling.

The operator relies on this information following a LOCA to secure HPI and throttle LPI, following a SBLOCA to throttle HPI and begin forced HPI cooling if

(continued)

BASES

LCO

16. Core Exit Temperature (continued)

needed, and following a MSLB and SG Tube Rupture to throttle HPI and isolate the affected SG.

There are a total of 52 Core Exit Thermocouples (CETs) per Oconee Unit. Twenty-four (12 per train) meet seismic and environmental qualification requirements (Category 1). The unit computer is the primary display for all 52 CETs. The CETs are distributed to provide monitoring of four or more in each quadrant for each train. The ICCM plasma displays (1 per train) located in the Control Room serve as safety related backup displays for the twenty-four Category 1 CETs. The range of the readouts is 50°F to 2300°F.

The ICCM CET function uses inputs from twelve incore thermocouples per train to calculate and display temperatures of the reactor coolant as it exits the core and to provide indication of thermal conditions across the core at the core exit. Each of the twelve qualified thermocouples per train is displayed on a spatially oriented core map on the plasma display. Trending of CET temperature is available continuously on the plasma display. The average of the five hottest CETs is trendable for the past forty minutes.

An evaluation was made of the minimum number of valid core exit thermocouples (CETs) necessary for inadequate core cooling detection. The evaluation determined the reduced complement of CETs necessary to detect initial core recovery and to trend the ensuing core heatup. The evaluations account for core nonuniformities and cold leg injection. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two sets of five valid CETs.

Table 3.3.8-1 Note (d) indicates that the subcooling margin monitor takes the average of the five highest CETs for each of the ICCM trains. Two channels ensure that a single failure will not disable the ability to determine the representative core exit temperature.

(continued)

BASES

LCO (continued)

17. Subcooling Monitor

The Subcooling Monitor is a Type A, Category 1 variable provided for verification and long term surveillance of core cooling. This variable is a computer calculated value using various inputs from the Primary System.

Two channels of indication are provided. One channel monitors RCS Loop A and the Core Saturation margin while another separate channel monitors RCS Loop B and the Core Saturation margin. The indication readouts are located in the control room. This variable also inputs to the unit computer through isolation buffers and is available for trend recording upon operator demand. The range of the readouts is 200°F subcooled to 50°F superheat. The control room display is through the ICCM plasma display unit.

A backup method for determining subcooling margin ensures the capability to accurately monitor RCS subcooling margin (Refer to Specification 5.5.17).

18. HPI System Flow

HPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for short term cooling requirements, to prevent HPI pump runout and inadequate NPSH, and to indicate the need for flow cross connect. HPI flow is throttled based on RCS pressure, subcooled margin, and pressurizer level. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two HPI trains. The channels provide flow indication over a range of 0 to 750 gpm.

19. LPI System Flow

LPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements, to prevent LPI pump runout and for flow balance. The indication is also used to identify an LPI pump operating at system pressures above its shutoff head. Flow measurement is

(continued)

BASES

LCO

19. LPI System Flow (continued)

provided by one channel per train with readout on an indicator and recorder. There are two LPI trains. The LPI channels provide flow indication over a range of 0 to 6000 gpm.

20. Reactor Building Spray Flow

Reactor Building Spray Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements and iodine removal and to prevent Reactor Building Spray and LPI pump runout. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two RBS trains. The channels provide flow indication over a range from 0 to 2000 gpm.

21. Emergency Feedwater Flow

EFW Flow instrumentation is a Type D, Category 1 variable provided to monitor operation of RCS heat removal via the SGs. Two channels provide indication of EFW Flow to each SG over a range of approximately 100 gpm to 1200 gpm. Redundant monitoring capability is provided by the two independent channels of instrumentation for each SG. Each pressure transmitter provides an input to a control room indicator. One channel also provides input to a recorder.

EFW Flow is the primary indication used by the operator to verify that the EFW System is delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate accidents and transients. The applicable accidents and transients are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6,

(continued)

BASES

APPLICABILITY (continued)

unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments.

Note 2 is added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.8-1. The Completion Time(s) of the inoperable channels of a Function are tracked separately for each Function starting from the time the Condition is entered for that Function.

A.1

When one or more Functions have one required channel inoperable, the inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience. This takes into account the remaining OPERABLE channel, the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval.

Condition A is modified by a Note indicating this Condition is not applicable to PAM Functions 14, 18, 19, and 20.

B.1

Required Action B.1 specifies initiation of action described in Specification 5.6.6 that requires a written report to be submitted to the NRC. This report discusses the results of

(continued)

BASES

ACTION

B.1 (continued)

the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. The Completion Time of "Immediately" for Required Action B.1 ensures the requirements of Specification 5.6.6 are initiated.

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. This Condition does not apply to the hydrogen monitor channels. The Completion Time of 7 days is based on the relatively low probability of an event requiring PAM instrumentation action operation and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

Condition C is modified by a Note indicating this Condition is not applicable to PAM Functions 10, 14, 18, 19, and 20.

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This action restores the monitoring capability of the hydrogen monitor. The 72 hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring. Continuous operation with two required channels inoperable is not acceptable because alternate indications are not available.

(continued)

BASES

ACTIONS

D.1 (continued)

Condition D is modified by a Note indicating this Condition is only applicable to PAM Function 10.

E.1

When one required BWST water level channel is inoperable, Required Action E.1 requires the channel to be restored to OPERABLE status. The 24 hour Completion Time is based on the relatively low probability of an event requiring BWST water and the availability of the remaining BWST water level channel. Continuous operation with one of the two required channels inoperable is not acceptable because alternate indications are not available. This indication is crucial in determining when the water source for ECCS should be swapped from the BWST to the reactor building sump.

Condition E is modified by a Note indicating this Condition is only applicable to PAM Function 14.

F.1

When a flow instrument channel is inoperable, Required Action F.1 requires the affected HPI, LPI, or RBS train to be declared inoperable and the requirements of LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 apply. The required Completion Time for declaring the train(s) inoperable is immediately. Therefore, LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 is entered immediately, and the Required Actions in the LCOs apply without delay. This action is necessary since there is no alternate flow indication available and these flow indications are key in ensuring each train is capable of performing its function following an accident. HPI, LPI, and RBS train OPERABILITY assumes that the associated PAM flow instrument is OPERABLE because this indication is used to throttle flow during an accident and assure runout limits are not exceeded or to ensure the associated pumps do not exceed NPSH requirements.

Condition F is modified by a Note indicating this Condition is only applicable to PAM Functions 18, 19, and 20.

(continued)

BASES

ACTION

G.1 (continued)

Required Action G.1 directs entry into the appropriate Condition referenced in Table 3.3.8-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met the Required Action and associated Completion Time of Condition C, D, or E, as applicable, Condition G is entered for that channel and provides for transfer to the appropriate subsequent Condition.

H.1 and H.2

If the Required Action and associated Completion Time of Conditions C, D or E are not met and Table 3.3.8-1 directs entry into Condition H, the unit must be brought to a MODE in which the requirements of this LCO do not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours and MODE 4 within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

I.1

If the Required Action and associated Completion Time of Condition C, D or E are not met and Table 3.3.8-1 directs entry into Condition I, alternate means of monitoring the parameter should be applied and the Required Action is not to shut down the unit, but rather to follow the directions of Specification 5.6.6 in the Administrative Controls section of the Technical Specifications. These alternative means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allowed time. The report provided to the NRC should discuss the alternative means used, describe the degree to which the alternative means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

Both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling capability. The subcooled margin monitors (SMM), and core-exit

(continued)

BASES

ACTIONS

I.1 (continued)

thermocouples (CET) provide an alternate means of monitoring for this purpose. The function of the ICC instrumentation is to increase the ability of the unit operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory.

The alternate means of monitoring the Reactor Building Area Radiation (High Range) consist of a combination of installed area radiation monitors and portable instrumentation.

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs apply to each PAM instrumentation Function in Table 3.3.8-1 except where indicated.

SR 3.3.8.1

Performance of the CHANNEL CHECK once every 31 days for each required instrumentation channel that is normally energized ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

The Frequency is based on operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal but more frequent checks of channels during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.8.2 and SR 3.3.8.3

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. This test verifies the channel responds to measured parameters within the necessary range and accuracy.

Note 1 to SR 3.3.8.3 clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.

For the Containment Area Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source.

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors or Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 and SR 3.3.8.3 (continued)

SR 3.3.8.2 is modified by a Note indicating that it is applicable only to Functions 7 and 10. SR 3.3.8.3 is modified by Note 2 indicating that it is not applicable to Functions 7 and 10. The Frequency of each SR is based on operating experience and is justified by the assumption of the specified calibration interval in the determination of the magnitude of equipment drift.

REFERENCES

1. Duke Power Company letter from Hal B. Tucker to Harold M. Denton (NRC) dated September 28, 1984.

2. UFSAR, Section 7.5.

3. NRC Letter from Helen N. Pastis to H. B. Tucker, "Emergency Response Capability - Conformance to Regulatory Guide 1.97," dated March 15, 1988.

4. Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983.

5. NUREG-0737, "Clarification of TMI Action Plan Requirements," 1980.

6. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality at lower power levels than can be seen on the wide range neutron flux instrumentation. These channels also provide the operator with a flux indication that reveals changes in reactivity and helps to verify that SDM is being maintained.

The source range instrumentation has four redundant count rate channels originating in four fission chambers. Four source range detectors are externally located symmetrically around the core. These channels are used over a counting range of 0.1 cps to 1E5 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.1 decades to +7 decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel.

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels are necessary to monitor core reactivity changes. They are the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

The source range neutron flux channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 1).

LCO

Two source range neutron flux channels shall be OPERABLE to provide the operator with redundant source range neutron instrumentation. The source range instrumentation provides

(continued)

BASES

LCO (continued)

the primary power indication at low power levels < 4E-4% RTP on wide range instrumentation and must remain OPERABLE for the operator to continue increasing power.

APPLICABILITY

Two source range neutron flux channels shall be OPERABLE in MODE 2 to provide redundant indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the wide range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1.

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring neutron flux and to provide an early indication of reactivity changes.

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2, "Nuclear Instrumentation."

ACTIONS

A.1

The Required Action for one required channel of the source range neutron flux indication inoperable with THERMAL POWER ≤ 4E-4% RTP on the wide range neutron flux instrumentation is to delay increasing reactor power until the channel is repaired and restored to OPERABLE status. This limits power increases in the range where the operators rely solely on the source range instrumentation for power indication. The Completion Time ensures the source range is available prior to further power increases. Furthermore, it ensures that power remains below the point where the wide range channels provide primary protection.

B.1, B.2, B.3, and B.4

With both required source range neutron flux channels inoperable with THERMAL POWER ≤ 4E-4% RTP on the wide range neutron flux instrumentation, the operators must take actions to limit the possibilities for adding positive reactivity. This is done by immediately suspending positive

(continued)

BASES

ACTIONS

B.1, B.2, B.3, and B.4 (continued)

reactivity additions, initiating action to insert all CONTROL RODS, and opening the control rod drive trip breakers within 1 hour. Periodic SDM verification is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than CONTROL ROD withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in this condition, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required Action B.1, Required Action B.2, and Required Action B.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action B.3 and Required Action B.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification provides reasonable assurance that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged.

C.1

With reactor power > 4E-4% RTP in MODE 2, 3, 4, or 5 on the wide range neutron flux instrumentation, continued operation is allowed with one or more required source range neutron flux channels inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the channel(s) to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until channels are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1 (continued)

channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction.

The Frequency, equivalent to every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels. When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant source range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with wide range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.9.2

For source range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels from the preamplifier input to the indicators. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of 18 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an 18 month interval, such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.10 Wide Range Neutron Flux

BASES

BACKGROUND

The wide range neutron flux channels provide the operator with an indication of reactor power from 1E - 8 to 200% of RTP and fully overlap the source and power range channels providing continuity of information needed during startup.

The wide range instrumentation has four log N channels originating in four electrically identical fission chambers. Each channel provides ten decades of flux level information in terms of the log of chamber count rate and startup rate. The startup rate which measures the rate of change of the neutron flux level, is displayed for the operator in a range from -0.1 decades to +7 decades per minute. A high startup rate of +2 decades per minute in either channel will initiate a control rod withdrawal inhibit.

APPLICABLE SAFETY ANALYSES

Wide range neutron flux channels are necessary to monitor core reactivity changes and are the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions.

The wide range neutron flux channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 1).

LCO

Two wide range neutron flux instrumentation channels shall be OPERABLE to provide the operator with redundant neutron flux indication. These enable operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling neutron flux transients that could result in reactor trip during power escalation.

(continued)

BASES (continued)

APPLICABILITY

The wide range neutron flux channels shall be OPERABLE in MODE 2 and in MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

The wide range instrumentation is designed to detect power changes during initial criticality and power escalation when the power range and source range instrumentation cannot provide reliable indications. Since these conditions can exist in, or propagate from, all of these MODES, the wide range instrumentation must be OPERABLE.

ACTIONS

A.1

If one required wide range channel becomes inoperable, the unit is exposed to the possibility that a single failure will disable all neutron monitoring instrumentation. To avoid this, the inoperable channel must be repaired or power must be reduced to the point where source range channels can provide neutron flux indication. Completion of Required Action A.1 places the unit in this state, and LCO 3.3.9, "Source Range Neutron Flux," requires OPERABILITY of two source range channels once this state is reached. If the one channel failure occurs when indicated power is < 4E-4% RTP, the Required Action prohibits increases in power above the source range capability.

The 2 hour Completion Time allows controlled reduction of power into the source range and is based on unit operating experience that demonstrates the improbability of the second wide range channel failing during the allowed interval.

B.1 and B.2

With two required wide range neutron flux channels inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the reactor in the next lowest condition for which the wide range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. The Completion Times are based on unit

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, equivalent to every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant wide range may not be available for comparison. CHANNEL

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1 (continued)

CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE wide range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

For wide range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an 18 month interval such that the instrument is not adversely affected by drift.

SR 3.3.10.3

SR 3.3.10.3 is the verification once each reactor startup of one decade of overlap with the source range neutron flux instrumentation. The wide range detector should be on scale and indicating $\geq$ 1E-8% of RTP when the source range detector is indicating $\leq 10^4$ counts per second in order for the wide range detector to indicate a one decade change prior to the source range detector going off scale. This ensures a continuous source of power indication during the approach to criticality.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.3.10.3 (continued) The test may be omitted if performed within the previous 7 days based on operating experience, which shows that source range and wide range instrument overlap does not change appreciably within this test interval.
REFERENCES	1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.11 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Instrumentation

BASES

BACKGROUND

The MSLB Detection and MFW Isolation instrumentation is designed to address containment overpressurization concerns by isolating main feedwater (MFW) to both steam generators during an MSLB and to mitigate core overcooling concerns.

Steam generator header pressure is used as input signals to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The turbine-driven emergency feedwater (TDEFW) pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. These functions are credited for mitigating an MSLB. The function of closing the main and startup feedwater block valves is not credited in the MSLB analysis for mitigation of containment overpressurization during a MSLB. However, the MSLB detection and MFW isolation circuitry performs this function.

There are three pressure transmitters per steam generator with each feeding a steam pressure signal to a signal isolator (when used) and bistable. These bistables are calibrated to provide an ON/OFF signal at the desired setpoint for actuation of the feedwater isolation circuitry. A pressure transmitter and its associated signal isolator(s) and bistable(s) constitute a MSLB detection analog channel.

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the digital channels will not result in an in advertent feedwater isolation. If either digital channel is actuated, an MFW isolation will occur. Energizing the master relay results in closure of contacts in various

(continued)

BASES

BACKGROUND (continued)

control circuits for systems and components used for the MSLB containment overpressurization protection. Therefore, when the master relay is energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. The two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, time delay, and test/manual actuation pushbutton are considered a feedwater isolation digital channel.

The feedwater isolation digital channels are enabled and disabled administratively rather than automatically. Appropriate operating procedures contain provisions to enable/disable the digital channels.

APPLICABLE SAFETY ANALYSES

Based on the containment pressure response reanalysis, the containment design pressure would be exceeded for a MSLB inside containment without operator action to isolate main feedwater and installed equipment necessary to automatically isolate main feedwater to both steam generators during a MSLB.

Steam generator header pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs are closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. All of these functions are credited for mitigating a MSLB inside containment.

The MSLB Detection and MFW Isolation Instrumentation satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3).

(continued)

BASES (continued)

LCO

This LCO requires that instrumentation necessary to initiate a MFW isolation shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the Function.

Three channels per SG are required to be OPERABLE to ensure that no single failure prevents MFW isolation. Each MSLB Detection and MFW Isolation instrumentation channel includes the sensor and measurement channel.

APPLICABILITY

The MSLB Detection and MFW Isolation Function shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure $\geq$ 700 psig because the SG inventory can be at a high energy level and contribute significantly to the peak pressure with a secondary side break. The main feedwater must be able to be isolated on each SG to limit mass and energy releases to the reactor building. Once the SG pressures have decreased below 700 psig, the MFW Isolation Function can be bypassed to avoid actuation during normal unit cooldowns. Also during MODE 3 the MFW isolation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since the function of the instrumentation is already fulfilled. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent. In MODES 4, 5, and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy and MSLB Detection and MFW Isolation instrumentation is not required to be OPERABLE.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or any of the transmitter or signal processing electronics, are found inoperable, then the Function provided by that channel must be declared inoperable and the unit must enter the appropriate Conditions.

A Note has been added to the ACTIONS indicating that a separate Condition entry is allowed for instrumentation channels associated with each SG (MFW isolation function).

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies to failures of a single MSLB Detection and MFW Isolation instrumentation channel in one or more MFW Isolation Functions.

With one channel inoperable in one or more MSLB Detection and MFW Isolation Function, the channel(s) must be placed in trip within 4 hours. Tripping the affected channel places the Function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the MSLB Detection and MFW Isolation Function is capable of performing its isolation function in the presence of any single random failure. The Completion Time of 4 hours is adequate to perform Required Action A.1.

B.1, B.2.1, and B.2.2

With two channels in one or more MSLB Detection and MFW Isolation Function inoperable or the Required Action and associated Completion Time of Condition A not met, the unit must be placed in MODE 3 within 12 hours and main steam header pressure must be reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1 (continued)

between each CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

Agreement criteria are based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified, where practical, to be reading at the bottom of the range and not failed downscale.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO required channels.

SR 3.3.11.2

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.3.11.2 (continued) The Frequency is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.
REFERENCES	1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.12 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Manual Initiation

BASES

BACKGROUND

The MSLB Detection and MFW Isolation manual initiation capability provides the operator with the capability to actuate the isolation function from the control room. This Function is provided in the event the operator determines that the Function is needed and does not automatically actuate. This is a backup Function to the automatic MFW isolation.

The MSLB Detection and MFW Isolation manual initiation circuitry satisfies the manual initiation and single-failure criterion requirements of IEEE-279-1971 (Ref. 1).

APPLICABLE SAFETY ANALYSES

The MFW Isolation Function credited in the safety analysis is automatic. However, the manual initiation Function is required by design as backup to the automatic Function and allows operators to actuate MFW Isolation whenever the Function is needed. Furthermore, the manual initiation of MFW Isolation may be specified in unit operating procedures.

The MSLB Detection and MFW Isolation manual initiation function satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

One manual initiation switch per actuation channel (A and B) is required to be OPERABLE. The MFW Isolation function, has two actuation or "trip" channels, channels A and B. Within each channel actuation logic there is one manual trip switch. When the manual switch is depressed, a full trip of actuation channel A or B occurs.

APPLICABILITY

The MFW Isolation manual initiation Function shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure $\geq$ 700 psig because SG inventory can be at a sufficiently high energy level to contribute significantly to the peak containment pressure with a secondary side

(continued)

BASES

APPLICABILITY (continued)	break. During MODE 3, the MFW Isolation manual initiation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled. In MODES 4, 5, and 6, the SG energy level is low and secondary side feedwater flow rate is low or nonexistent.

ACTIONS

A.1

With one manual initiation switch inoperable, the manual initiation switch must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and MFW Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and MFW Isolation Function.

B.1

With both manual initiation switches inoperable or the Required Action and associated Completion Time of Condition A not met, the unit must be placed in MODE 3 within 12 hours and the main steam header pressure reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to

(continued)

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.3.12.1</u> (continued) perform its safety function, while the risks of testing during unit operation is avoided.
REFERENCES	1. IEEE-279-1971, April 1972. 2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.13 Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation Logic Channels

BASES

BACKGROUND

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the digital channels will not result in an inadvertent feedwater isolation. If either digital channel is actuated, an MFW isolation will occur. Energizing the master relay results in closure of contacts in various control circuits for systems and components used for the MSLB containment overpressurization protection. Therefore, when the master relay is energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. Each of the two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, and time delay is considered a feedwater isolation digital channel.

APPLICABLE SAFETY ANALYSES

MSLB circuitry is installed equipment necessary to automatically isolate main feedwater to both steam generators during a MSLB.

Steam generator outlet pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for heat removal. All of these functions are credited for mitigating a MSLB inside containment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	The MSLB Detection and MFW Isolation logic channels satisfy Criterion 3 of 10 CFR 50.36 (Ref. 1).
LCO	Two channels of MSLB Detection and MFW Isolation automatic actuation logic shall be OPERABLE. There are only two channels of automatic actuation logic. Therefore, violation of this LCO could result in a complete loss of the automatic Function assuming a single failure of the other channel.
APPLICABILITY	The MSLB Detection and MFW Isolation automatic actuation logic channels shall be OPERABLE in MODES 1 and 2, and MODE 3 with main steam header pressure $\geq$ 700 psig because SG inventory can be at a high energy level and can contribute significantly to the peak containment pressure during a secondary side line break. Also, during MODE 3, the MFW Isolation function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent.
ACTIONS	A.1 With one automatic actuation logic channel inoperable, the channel must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and MFW Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and MFW Isolation Function.

(continued)

BASES

ACTIONS (continued)

B.1, B.2.1, and B.2.2

With both logic channels inoperable or the Required Action and associated Completion Time not met, the unit must be placed in MODE 3 within 12 hours and the main steam header pressure must be reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. This test verifies MFW Isolation automatic actuation logics are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during unit operation is avoided.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.14 Emergency Feedwater (EFW) Pump Initiation Circuitry

BASES

BACKGROUND

EFW pump initiation circuitry is designed to provide safety grade means of controlling the secondary system as a heat sink for core decay heat removal. To ensure the secondary system remains a heat sink, the EFW pump initiation circuitry takes action to initiate EFW when the primary source of feedwater is lost. These actions ensure that a source of cooling water is available to be supplied to a steam generator (SG), thereby establishing the heat sink temperature at the saturation temperature of the secondary system.

EFW is initiated to restore a source of cooling water to the secondary system when conditions indicate that the normal source of feedwater is insufficient to continue heat removal. The EFW pump initiation circuitry contains devices that generate an EFW pump initiation signal when loss of main feedwater pumps are indicated by low hydraulic oil pressure. Each EFW Pump initiation circuit is fed by two loss of main feedwater (LOMF) instrumentation channels (hydraulic oil pressure switches) common only to that circuit which feed a two-out-of-two logic circuit that automatically starts each EFW pump. Each EFW pump also has a dedicated manual start circuit.

Each motor driven EFW pump is normally controlled by a four-position, OFF-AUTO1-AUTO2-RUN, control switch located in the control room. The pump can be manually started by turning the control switch to the RUN position. In the AUTO1 mode, each motor-driven EFW pump starts automatically after a sustained low water level in either steam generator for greater than 30 seconds. In the AUTO2 Mode, each pump starts automatically on low steam generator level or loss of both main feedwater pumps.

The turbine-driven EFW pump is started by opening valve MS-93 which admits steam to the pump turbine. A four-position, RUN-AUTO-OFF-PULL TO LOCK, control switch is provided to control operation of MS-93. The switch is maintained in the AUTO position. In the AUTO mode, MS-93 opens on low hydraulic oil pressure in both MFW pumps.

(continued)

BASES

BACKGROUND (continued)

When the switch is in the RUN position, MS-93 is opened.

Loss of both MFW Pumps was chosen as an EFW automatic initiating parameter because it is a direct and immediate indicator of loss of MFW.

EFW is also initiated by a low level in the SG (after a 30 second delay to prevent spurious actuation) for SG dryout protection. EFW initiation for SG dryout protection is not required by this Specification. Finally, EFW is also initiated by a loss of both MFW pumps as indicated by low hydraulic oil pressure as part of the ATWS Mitigation Circuitry (AMSAC), which is a system provided to comply with the requirements to reduce risk from an anticipated transient without scram (ATWS). EFW initiation for ATWS mitigation is not required by this Specification.

APPLICABLE SAFETY ANALYSES

The transient which forms the basis for initiation of the EFW systems is a loss of MFW transient. In the analysis of the transient, MFW pump turbine low control oil pressure is the parameter assumed to automatically initiate EFW.

The EFW pump initiation circuitry satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two loss of main feedwater (LOMF) pump instrumentation channels and an automatic initiation circuit and a manual initiation circuit are required OPERABLE for each EFW pump. Each LOMF instrumentation channel is considered to include the sensors and measurement channels. The LCO is modified by a Note that limits the OPERABILITY required for the automatic initiation circuitry to MODES 1 and 2.

APPLICABILITY

The initiation circuitry for EFW pumps shall be OPERABLE in MODES 1, 2 and 3 and in MODE 4 when the steam generator is relied upon for heat removal. In MODE 4 when the steam generator is not relied upon for heat removal, and MODES 5, and 6, the primary system temperatures are too low to allow

(continued)

BASES

APPLICABILITY (continued)

the SGs to effectively remove energy and EFW Pump initiation instrumentation is not required to be OPERABLE.

ACTIONS

The ACTIONS are modified by a Note indicating that this Specification may be entered independently for each EFW pump initiation circuit. The Completion Time(s) of the inoperable channels for each EFW automatic initiation circuit are tracked separately for each circuit starting from the time the Condition is entered for that circuit.

A.1

With one or more required EFW pump initiation circuits with one LOMF channel inoperable, the channel(s) must be placed in trip within 1 hour. With the channel in trip, the resultant logic is one-out-of-one. This channel may be considered placed in trip, after tripping, by installing jumpers or by other means that assure the channel remains in the tripped condition.

B.1

With one or more EFW pump initiation circuits inoperable or the Required Action and associated Completion Time of Condition A not met, the affected EFW pump(s) must be declared inoperable immediately since the initiation function is no longer capable of performing its safety function.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

A CHANNEL FUNCTIONAL TEST verifies the function of the required trip of the channel. Setpoints for trip must be found within the Allowable Value. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1 (continued)

The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.14.2

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The Frequency is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapters 7 and 15.

2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a high energy line break (HELB).

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a break in this section of main steam header has a very low probability. The main MSLB without ICS and without operator action is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

The TSVs remain open during power operation. These valves close upon a reactor trip.

a. For an HELB or an MSLB inside containment, the analysis assumes the TSV in the affected steam generator remains open. For this scenario, steam is discharged into containment from both steam generators until closure of the TSV in the intact steam generator occurs. After TSV closure, steam is discharged into containment only from the affected steam generator.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. An event such as increased steam flow through the turbine will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSV Closure function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two TSV Closure channels are required to be OPERABLE.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 3).

(continued)

BASES (continued)

APPLICABILITY

Both TSV Closure channels must be OPERABLE in MODES 1, 2 and 3 with any TSVs open. In these conditions when there is significant mass and energy in the RCS and steam generators, the TSV Closure function must be OPERABLE or the TSVs closed. When the TSVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low. Therefore, the TSV Closure channels are not required to be OPERABLE. In MODES 5 and 6, the steam generators do not contain a significant amount of energy because their temperature is below the boiling point of water; therefore, the TSV Closure channels are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one or more TSV Closure channels inoperable, all TSVs must be declared inoperable. A Completion Time of 1 hour is provided to return the TSV Closure channels to OPERABLE status. The 1 hour Completion Time is sufficient time to correct minor problems.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended function. This test verifies the TSV Closure automatic actuation channels are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic channels. The test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of TSVs during testing at power. The Frequency of 31 days is based on engineering judgment and operating experience, which determined the interval provided adequate confidence that the TSV Closure channels are available to perform their safety function, while the risks of testing at operation are avoided.

(continued)

BASES (continued)

REFERENCES	1.	UFSAR, Section 15.13.
	2.	10 CFR 50.36.
	3.	10 CFR 100.

B 3.3 INSTRUMENTATION

B 3.3.16 Reactor Building (RB) Purge Isolation—High Radiation

BASES

BACKGROUND

The RB Purge Isolation—High Radiation Function closes the RB purge valves. This action isolates the RB atmosphere from the environment to minimize releases of radioactivity in the event an accident occurs.

The radiation monitoring system measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The LCO addresses only the gas sampler portion of this system (RIA-45).

The trip setpoint is chosen sufficiently below hazardous radiation levels to ensure that the consequences of an accident will be acceptable, provided the unit is operated within the LCOs at the onset of an accident or transient and the equipment functions as designed.

The closure of the purge valves ensures the RB remains as a barrier to fission product release. There is no bypass for this function.

APPLICABLE SAFETY ANALYSES

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). A minimum fuel transfer canal water level and the minimum decay time of 72 hours prior to movement of irradiated fuel assemblies from the reactor ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the guideline values specified in 10 CFR 100. The design basis for fuel handling accidents has historically separated the radiological consequences from the containment capability. The NRC staff has treated the containment capability for fuel handling conditions as a logical part of the "primary success path" to mitigate fuel handling accidents, regardless of the assumptions used to

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

calculate the radiological consequences of such accidents (Ref. 1).

The RB Purge Isolation System satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

One channel of RB Purge Isolation - High Radiation instrumentation is required to be OPERABLE. OPERABILITY of the instrumentation includes proper operation of the sample pump. This LCO addresses only the gas sampler portion of the System.

APPLICABILITY

The RB purge isolation — high radiation instrumentation shall be OPERABLE whenever CORE ALTERATIONS or movement of irradiated fuel assemblies within the RB is taking place. These conditions are those under which the potential for fuel damage, and thus radiation release, is the greatest. While in MODES 1, 2, 3, and 4, the Purge Valve Isolation System does not need to be OPERABLE because the purge valves are required to be sealed closed. While in MODES 5 and 6, without fuel handling in progress, the Purge Valve Isolation System does not need to be OPERABLE because the potential for a radioactive release is minimized. The need to use the purge valves in MODES 5 and 6 is in preparation for entry. This capability is required to minimize doses for personnel entering the building and is independent of the automatic isolation capability.

ACTIONS

A.1, A.2.1, and A.2.2

Condition A applies to failure of the high radiation purge function during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB.

With one channel inoperable during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB, the RB purge valves must be closed, or CORE ALTERATIONS and movement of irradiated fuel assemblies within the RB must be suspended. Required Action A.1 accomplishes the function of the high radiation channel. Required Action A.2.1 and

(continued)

BASES

ACTIONS

A.1, A.2.1, and A.2.2 (continued)

Required Action A.2.2 place the unit in a configuration in which purge isolation on high radiation is not required. The Completion Time of "Immediately" is consistent with the urgency associated with the loss of RB isolation capability under conditions in which the fuel handling accidents are possible and the high radiation function provides the only automatic actions to mitigate radiation release.

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1

SR 3.3.16.1 is the performance of the CHANNEL CHECK for the RB purge isolation – high radiation instrumentation once every 12 hours to ensure that a gross failure of instrumentation has not occurred. The CHANNEL CHECK is normally a comparison of the parameter indicated on the radiation monitoring instrumentation channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps to ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. The 12 hour Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Additionally, control

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1 (continued)

room alarms and annunciators are provided to alert the operator to various "trouble" conditions associated with the instrument.

SR 3.3.16.2

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channel can perform its intended function. The frequency requires the isolation capability of the reactor building purge valves to be verified functional once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. This ensures that this function is verified prior to irradiated fuel assembly handling within containment. This test verifies the capability of the instrumentation to provide the RB isolation.

SR 3.3.16.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The 18 month Frequency is based on engineering judgment and industry accepted practice.

REFERENCES

1. NRC Letter to RG & E dated December 7, 1995 R.E. Ginna Nuclear Power Plant conversion to Improved Standard Technical Specifications - Resolution of Ginna Design Basis for Refueling Accidents.

2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.17 Emergency Power Switching Logic (EPSL) Automatic Transfer Function

BASES

BACKGROUND

The transfer circuits of the EPSL are designed with sufficient redundancy to assure that power is supplied to the unit Main Feeder Buses (MFBs) and, hence, to the unit's essential loads, under accident conditions. The logic system monitors the normal and emergency power sources and, upon loss of the normal power source (the unit auxiliary transformer), the logic seeks an available alternate source of power.

The Load Shed and Transfer to Standby Circuits are designed to energize the MFBs from the Standby Buses powered from either Keowee or Lee when voltage is lost or is insufficient from the Normal and Startup sources. The Load Shed signal is generated to separate nonessential loads from the MFBs to ensure the CT-4 or CT-5 transformers supplying the Standby Buses are not overloaded. The Load Shed timers and Transfer to Standby timers are set such that, if no power is available from the startup source for approximately 11 seconds, the startup source breakers are prohibited from closing and the standby bus to MFB breakers receive a permissive to close.

The Retransfer to Startup logic provides the emergency power switching logic the capability to retransfer essential loads from the Standby Bus to the startup source, if available, should power to both standby buses be lost for more than 5 seconds.

The EPSL automatic transfer function is designed to perform their function assuming a single failure. There are two automatic transfer channels, with one channel consisting of Channel A of the Load Shed and Transfer to Standby function and Channel A of the Retransfer to Startup function and the other consisting of Channel B of both of these functions.

APPLICABLE SAFETY ANALYSES

The EPSL Automatic Transfer function is required for the engineered safeguards (ES) equipment to function in any accident with a loss of offsite power.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The limiting accident for the EPSL transfer functions is a LOCA with a simultaneous loss of offsite power (Ref. 1). The loss of offsite power is considered to occur coincident with ES actuation. In this scenario, the Load Shed and Transfer to Standby function reenergizes the affected unit's MFBs from the standby buses which are powered from Keowee or Lee.

The analyses assume that the maximum time the MFBs will be deenergized is 33 seconds. This time is derived from the 48 second time requirement for ECCS injection minus the 15 second ECCS valve stroke time requirement.

EPSL automatic transfer functions are part of the primary success path and function to mitigate an accident or transient that presents a challenge to the integrity of a fission product barrier. The EPSL automatic transfer function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two channels of the Automatic Transfer Function, with one channel consisting of Channel A of the Load Shed and Transfer to Standby function and Channel A of the Retransfer to Startup function and the other consisting of Channel B of both of these functions, are required to be OPERABLE. Failure of one channel reduces the reliability of the affected Functions.

The requirement for two channels to be OPERABLE ensures that one channel of the function will remain OPERABLE if a single failure has occurred. The remaining channel can perform the safety function.

APPLICABILITY

The automatic transfer function of EPSL is required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL automatic transfer function is not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

(continued)

BASES (continued)

ACTIONS

A.1

If one channel is inoperable, it must be restored to OPERABLE status within 24 hours. With one channel inoperable, the remaining channel is capable of providing necessary transfer functions to ensure power is provided to the MFBs. The 24 hour Completion Time is considered appropriate based on engineering judgement, taking into consideration the time required to complete the required action.

B.1 and B.2

With the Required Action and associated Completion Time not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 in 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to allow for a controlled shutdown.

SURVEILLANCE REQUIREMENTS

SR 3.3.17.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the EPSL automatic transfer function. The ES inputs to the Load Shed and Transfer to Standby function and the Retransfer to Startup function are verified to operate properly during an automatic transfer of the Main Feeder Buses to the Startup Transformer, Standby Buses, and retransfer to the Startup Transformers. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapters 6 and 15.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.18 Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits

BASES

BACKGROUND

The EPSL voltage sensing circuits for the Startup Transformer, Standby Bus #1, Standby Bus #2, and the Auxiliary Transformer provide input to the EPSL controls to actuate breakers and initiate transfer control sequences. Each phase of each source has an individual potential transformer feeding a 2 out of 3 logic for determining the status of the power source. The voltage sensing circuits also provide trip signals to the breaker control circuitry for the normal incoming breakers (N breakers), startup incoming breakers (E breakers), and CT-5 incoming breakers (SL breakers).

The EPSL system is designed to ensure power is supplied to the main feeder buses (MFBs) during a LOCA. In order for it to perform this function, the voltage sensing circuits for the Startup Transformer, Auxiliary Transformer, Standby Bus #1, and Standby Bus #2 must be OPERABLE. These voltage sensing circuits provide input to the EPSL transfer functions. The transfer functions utilize the voltage sensing circuits to initiate breaker operations to ensure the MFBs are connected to an energized source (startup or standby). The N and E breakers also get direct trips from the two-out-of-three logic.

No protective relay lockouts or inhibits can be present to prevent the connection of required AC power source(s) to the MFBs from the Control Room because their presence will not allow closure of the associated breaker.

APPLICABLE SAFETY ANALYSES

The EPSL voltage sensing circuits are required for the engineered safeguards (ES) equipment to function in any accident with a loss of offsite power. The limiting accident for the EPSL voltage sensing circuits is a loss-of-coolant accident (LOCA) with a simultaneous loss of offsite power (Ref. 1).

The EPSL voltage sensing circuits satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2).

(continued)

BASES (continued)

LCO

Three channels of each EPSL voltage sensing circuit (Auxiliary Transformer, Startup Transformer, Standby Bus #1, Standby Bus #2) are required to be OPERABLE. These circuits and associated channels ensure that no single failure can cause a loss of required ES equipment.

The LCO is modified by two Notes. Note 1 removes Auxiliary Transformer voltage sensing requirements when both N breakers are open. The function of the Auxiliary Transformer Voltage Sensing circuits is to provide a trip signal to the N breakers. When the N breakers are open, the Auxiliary Transformer voltage sensing circuits are not required and, therefore, need not be OPERABLE. Note 2 requires only the EPSL voltage sensing circuits associated with required AC power source(s) to be OPERABLE when not in MODES 1, 2, 3, and 4.

APPLICABILITY

The EPSL voltage sensing circuits are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL voltage sensing circuits associated with required AC power source(s) required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems needed to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

(continued)

BASES (continued)

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Voltage Sensing Circuit.

A.1

If one required channel is inoperable in one or more voltage sensing circuits, it must be restored to OPERABLE status within 24 hours. With one channel inoperable, the remaining two channels are capable of providing the voltage sensing function. The 24 hour Completion Time is considered appropriate based on engineering judgement taking into consideration the time required to complete the required action.

B.1 and B.2

With the Required Action and associated Completion Time not met in MODES 1, 2, 3 and 4, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 in 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to allow for a controlled shutdown.

C.1 and C.2

With two or more channels of a required circuit inoperable when not in MODES 1, 2, 3, and 4 or the Required Action and associated Completion Time not met when not in MODES 1, 2, 3, and 4, the affected AC power sources(s) must be declared inoperable immediately. The appropriate Required Actions will be implemented in accordance with LCO 3.8.2, "AC Sources – Shutdown."

D.1

With the Required Action and associated Completion Time not met during movement of irradiated fuel assemblies, movement of fuel assemblies must be suspended immediately. Suspension does not preclude completion of actions to establish a safe conservative condition. This action minimizes the probability or the occurrence of postulated

(continued)

BASES

ACTIONS

D.1 (continued)

events. The Completion Time of immediately is consistent with the required times for actions requiring prompt attention.

SURVEILLANCE REQUIREMENTS

SR 3.3.18.1

A CHANNEL FUNCTIONAL TEST is performed on each voltage sensing circuit channel to ensure the channel will perform its function. A circuit is defined as three channels, one for each phase. Each channel consists of components from the sensing power transformer through the circuit auxiliary relays which operate contacts in the EPSL logic and breaker trip circuits. Minimum requirements consist of individual channel relay operation causing appropriate contact responses within associated loadshed/breaker circuits, alarm activations, and proper indications for the sensing circuit control power status. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

REFERENCES

1. UFSAR, Chapters 6 and 15.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.19 Emergency Power Switching Logic (EPSL) 230 kV Switchyard Degraded Grid Voltage Protection (DGVP)

BASES

BACKGROUND

Two levels of protection are provided to assure the degradation of voltage from offsite sources does not adversely impact the function of safety-related systems and components. The first level of protection is provided by the EPSL Degraded Grid Protection System (DGPS). The second level of protection is provided by undervoltage relaying on the E and N breakers (reference LCO 3.3.18, "EPSL Voltage Sensing Circuits") which protects from loss of voltage.

The DGPS, upon indication of inadequate voltage, provides an alarm to the Unit 1 & 2 Control Room. If an engineered safeguards (ES) Channel 1 or 2 signal from any unit is sensed by the DGPS, while the voltage is below acceptable levels, the DGPS will initiate an isolation of the 230 kV switchyard Yellow Bus to ensure the onsite overhead emergency power path is available. Each DGPS actuation logic channel is capable of initiating isolation of the overhead emergency power path. This ensures the startup transformers are not connected to a degraded source of power. In this event, ES loads are provided power from the standby buses.

Based on operating experience, degradation of voltage in the 230 kV switchyard does not last for an extended period of time. Administrative procedures are in place to assure timely actions are taken to restore the voltage.

There are three undervoltage relays installed to monitor the switchyard voltage, one on each phase (X, Y, Z) of the 230 kV Yellow Bus. The undervoltage relay contacts are arranged in a two-out-of-three logic sequence which feeds two redundant time delay relays. The time delay relays prevent spurious actuations, but still provide adequate response time for voltage transients. Either of the two redundant time-delay relays will cause either of the two sets of actuating relays to initiate switchyard isolation. Circuit control power is fed from the 230 kV Switchyard 125 VDC system.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The EPSL Degraded Grid Voltage Protection function is required to ensure adequate voltage is available during an ES actuation when system grid voltages are not adequate (Ref. 1). Based on calculations, 219 kV is the minimum switchyard voltage that will ensure proper operation of loads during ES actuation.

The EPSL Degraded Grid Voltage Protection satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Three degraded grid voltage sensing relay channels are required to be OPERABLE. Failure of one channel reduces the reliability of the function. The requirement for three channels to be OPERABLE ensures that two channels will remain OPERABLE if a failure has occurred in one channel. The remaining channels can perform the safety function.

Two channels of the Degraded Grid Voltage Protection Actuation Logic function are required to be OPERABLE. The switchyard isolation circuit is considered a part of this logic channel. Therefore, if a switchyard isolation channel is inoperable, then one DGVP actuation channel is inoperable. The requirement for two channels to be OPERABLE ensures that one channel will remain OPERABLE if a failure has occurred in one channel. The remaining channel can perform the safety function.

APPLICABILITY

The DGPS functions are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL DGVP functions are not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

(continued)

BASES (continued)

ACTIONS

A.1

If one DGVP voltage sensing channel is inoperable, the channel must be placed in trip within 72 hours. Tripping the affected channel places the function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the DGVP function is capable of performing its DGVP function in the presence of a single failure. With one channel inoperable, the remaining channels are capable of providing the DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation, and the probability of an event requiring ES operation.

B.1

If one DGVP actuation logic channel is inoperable, the actuation logic channel must be restored to OPERABLE status within 72 hours. With one actuation logic channel inoperable, the remaining actuation logic channel is capable of providing the DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation, and the probability of an event requiring ES operation.

C.1 and C.2

With two or more voltage sensing channels or both actuation logic channels inoperable or the Required Action and associated Completion Time of Condition C not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 in 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable based on operating experience and to allow for a controlled shutdown.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.19.1

A CHANNEL FUNCTIONAL TEST is performed on each DGVP voltage sensing channel and DGVP actuation logic channel to ensure the entire channel will perform its intended function. Any setpoint adjustments shall be consistent with the assumptions of the setpoint analysis. The CHANNEL FUNCTIONAL TEST of the DGVP actuation logic channels includes verifying actuation of the switchyard isolation circuitry. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

SR 3.3.19.2

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapter 8.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.20 Emergency Power Switching Logic (EPSL) CT-5 Degraded Grid Voltage Protection (DGVP)

BASES

BACKGROUND

Two levels of protection are provided for the standby buses to assure that degradation of voltage from the 100 kV transmission system does not adversely impact the function of safety related systems and components. The first level of protection is provided by the EPSL CT-5 Degraded Grid Protection System. The second level of protection is provided by undervoltage relaying on the standby buses (reference LCO 3.3.18, "EPSL Voltage Sensing Circuits") which protects from loss of voltage.

Three undervoltage sensing relays provide common input to two channels of actuating logic. In addition to the three phase undervoltage sensing relays, each channel includes one time-delay relay, one auxiliary relay, and one associated single phase undervoltage sensing relay. Each channel trip signal passes through a selector switch, which either allows or inhibits the trip signal, to actuate one trip coil in each SL breaker. Inoperability of any voltage sensing channel reduces the logic for the voltage sensing function to a two-out-of-two. Loss of two or more voltage sensing relays results in inoperability of both channels of actuation logic.

APPLICABLE SAFETY ANALYSES

The EPSL CT-5 Degraded Grid Voltage Protection function is required to ensure adequate voltage is available during an ES actuation concurrent with a loss of offsite power or degraded voltage from the 230 kV switchyard when ES loads are supplied by the standby buses (Ref.1). Based on calculations, 4.155 kV is the minimum voltage that will ensure proper operation of loads during ES actuation.

This system is only required to be OPERABLE when the unit is in MODES 1, 2, 3, and 4 and the standby buses are energized without being electrically separated from the grid and offsite loads. System design is to provide protection for ES components caused by voltage droop due to inrush as the unit connects to the standby buses. The system is not a substitute for the dedicated line from Lee Gas Turbines.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Lee Feeder breakers (SL) have no automatic close functions. However, this system does provide additional flexibility for the Station electrical system and operators in available power source options.

The EPSL CT-5 Degraded Grid Voltage Protection satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Three CT-5 degraded grid voltage sensing relay channels are required to be OPERABLE. Failure of one channel reduces the reliability of the function. The requirement for three channels to be OPERABLE ensures that two channels will remain OPERABLE if a failure has occurred in one channel. The remaining voltage sensing channels can perform the safety function.

Two channels of the CT-5 Degraded Grid Voltage Protection Actuation Logic function are required to be OPERABLE. The requirement for two channels to be OPERABLE ensures that one channel will remain OPERABLE if a failure has occurred in one channel. The remaining channel can perform the safety function.

APPLICABILITY

The CT-5 DGPS functions are required to be OPERABLE in MODES 1, 2, 3, and 4 when standby buses are energized without being electrically separated from grid or loads to ensure adequate voltage protection should a unit be transferred to the standby bus during an event requiring an ES actuation.

The EPSL CT-5 DGVP functions are not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

ACTIONS

A.1

If one CT-5 DGVP voltage sensing relay channel is inoperable, the channel must be placed in trip within 72 hours. Tripping the affected channel places the function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the DGVP

(continued)

BASES

ACTIONS

A.1 (continued)

function is capable of performing its DGVP function in the presence of any single random failure. With one channel inoperable, the remaining voltage sensing channels are capable of providing the DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation, and the probability of an event requiring an ES actuation.

B.1

If one CT-5 DGVP actuation logic channel is inoperable, the actuation logic channel must be restored to OPERABLE status within 72 hours. With one actuation logic channel inoperable, the remaining actuation logic channel is capable of providing the CT-5 DGVP function. The 72 hour completion time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation and the probability of an event requiring an ES actuation.

C.1 and C.2

If two or more voltage sensing relay channels or two actuation logic channels are inoperable, automatic protection from degraded grid voltage for the standby buses powered from the 100 kV transmission system is not available. Continued operation is allowed provided that the SL breakers are opened within one hour.

Additionally, with the Required Action and associated Completion Time of Condition A or B not met, the SL breakers must be opened within one hour. This arrangement provides a high degree of reliability for the emergency power system. The one hour Completion Time is based on engineering judgement taking into consideration the infrequency of actual grid system voltage degradation and the probability of an event requiring an ES actuation.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.20.1

A CHANNEL FUNCTIONAL TEST is performed on each CT-5 DGVP voltage sensing channel and each CT-5 DGVP actuation logic channel to ensure the entire channel will perform its intended function. Any setpoint adjustments shall be consistent with the assumptions of the setpoint analysis. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

SR 3.3.20.2

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The Frequency is justified by the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapter 8.

2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.21 Emergency Power Switching Logic (EPSL) Keowee Emergency Start Function

BASES

BACKGROUND

The Keowee Emergency Start function of EPSL provides a start signal to the two on-site emergency power sources and sets up controls for the emergency mode. There are two channels of the Emergency Start function. Each channel is capable of starting both Keowee units and activating the controls for the emergency mode.

The Emergency Start channels 1 and 2 are actuated from Engineered Safeguards channels 1 and 2 respectively. The Emergency Start channels can also be activated manually from each control room (i.e., two emergency start switches in the Unit 1 and 2 control room and two emergency start switches in the Unit 3 control room) or cable spread rooms. There are two independent channels associated with each Oconee unit.

During a loss-of-coolant accident (LOCA) with a simultaneous loss of offsite power, the Keowee Emergency Start function of EPSL sends a start signal to both Keowee units. Logic is also actuated that ensures separation of both Keowee units from the system grid. Connection of the Keowee Unit aligned to the overhead power path is allowed only after a separate logic sequence (indicating switchyard isolation logic is complete which is not associated with the Keowee Emergency Start function) verifies the yellow bus is separated from the grid.

The Keowee Emergency Start function also disables non critical protective interlocks and trips associated with the Keowee generators. This ensures the generators can remain available as an emergency power source despite minor failures or malfunctions.

The Keowee Emergency Start circuitry is designed such that no single failure can prevent an Emergency Start signal from reaching the Keowee units. Each channel is independent of the other and only one channel is required to perform the entire safety function.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The EPSL Keowee Emergency Start function is required for the engineered safeguards (ES) equipment to function in any accident with a loss of offsite power. The limiting accident for the EPSL voltage sensing circuits is a loss-of-coolant accident (LOCA) with a simultaneous loss of offsite power (Ref. 1).

The EPSL Keowee Emergency Start Function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two channels of the Keowee Emergency Start function are required to be OPERABLE. Failure of one channel reduces the reliability of the function.

The requirement for two channels to be OPERABLE ensures that one channel will remain OPERABLE if a failure has occurred. The remaining channel can perform the safety function.

APPLICABILITY

The EPSL Keowee Emergency Start function is required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that power is provided from AC Sources to the AC Distribution system within the time assumed in the accident analyses.

The EPSL Keowee Emergency Start function is not required to be OPERABLE in MODES 5 and 6 since more time is available for the operator to respond to a loss of power event.

ACTIONS

<u>A.1</u>

If one channel is inoperable, then a failure of the other channel could prevent starting the Keowee units. With one channel inoperable, the remaining channel is capable of providing the Keowee Emergency Start function. The 72 hour Completion Time is considered appropriate based on engineering judgement taking into consideration the time required to complete the required action.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

With the Required Action and associated Completion Time not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 in 12 hours and to MODE 5 within 84 hours. The allowed Completion Times are reasonable, based on operating experience, to allow for a controlled shutdown.

C.1

With both channels of the Keowee Emergency Start function inoperable then both Keowee Hydro Units must be declared inoperable immediately. The appropriate Required Actions will be implemented in accordance with LCO 3.8.1, "AC Sources – Operating."

SURVEILLANCE REQUIREMENTS

SR 3.3.21.1

A CHANNEL FUNCTIONAL TEST is performed on each Keowee Emergency Start channel to ensure the channel will perform its function during an automatic transfer of the Main Feeder Buses to the Startup Transfer, Standby Buses, and retransfer to the Startup Transformers. The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

REFERENCES

1. UFSAR, Chapters 6 and 15.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.22 Emergency Power Switching Logic (EPSL) Manual Keowee Emergency Start Function

BASES

BACKGROUND

The Keowee Emergency Start function of EPSL provides a start signal to the two on-site emergency power sources and sets up controls for the emergency mode. There are two channels of the Emergency Start function. Each channel is capable of starting both Keowee units and activating the controls for the emergency mode.

The Emergency Start channels 1 and 2 are actuated from Engineered Safeguards channels 1 and 2 respectively. The Emergency Start channels can also be activated manually from each control room (i.e., two emergency start switches in the Unit 1 and 2 control room and two emergency start switches in the Unit 3 control room) or cable spread rooms. There are two independent channels associated with each Oconee unit.

APPLICABLE SAFETY ANALYSES

The OPERABILITY of the Manual Keowee Emergency Start Function during MODES 5 and 6 and during movement of irradiated fuel assemblies ensures that adequate AC electrical power is provided to mitigate events postulated during shutdown, such as a fuel handling accident.

The EPSL Manual Keowee Emergency Start Function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 1).

LCO

One channel of the Manual Keowee Emergency Start function, consisting of a manual initiation switch and an Emergency Start channel, is required to be OPERABLE.

(continued)

BASES (continued)

APPLICABILITY

The Manual Keowee Emergency Start function required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provides assurance that:

a. Systems needed to provide adequate coolant inventory makeup are available for the irradiated fuel assemblies;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

ACTIONS

A.1

If the required Manual Keowee Emergency Start channel is inoperable, both Keowee Hydro Units must be declared inoperable immediately. Therefore LCO 3.8.2 is entered immediately, and the required Completion Times for the appropriate Required Actions apply without delay.

SURVEILLANCE REQUIREMENTS

SR 3.3.22.1

A CHANNEL FUNCTIONAL TEST is performed on the required Manual Keowee Emergency Start channel to ensure the channel will perform its function. The Frequency of 12 months is based on engineering judgment and operating experience that determined testing on a 12 month interval provides reasonable assurance that the circuitry is available to perform its safety function.

REFERENCES

1. 10 CFR 50.36.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.3 - INSTRUMENTATION

ATTACHMENT 3

CTS MARKUP AND DISCUSSION OF CHANGES

Specification 3.3.1

A1 < Except as marked >

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability

~~Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.~~

Objective

~~To provide automatic protective action to prevent any combination of process variables from exceeding a safety limit.~~

Specification

LCO AVs The reactor protective system trip setpoints and the permissible bypasses for the instrument channels shall be as stated in Table 2.3.1.

T 3.3.1-1 Item 7 AV The pump monitors shall produce a reactor trip when a loss of two pumps occurs and the reactor is at power operation greater than 2.0% of rated power.

Bases

A2

~~The reactor trip setpoints for reactor protective system (RPS) instrumentation are given in Table 2.3-1. The trip setpoints have been selected to ensure that the core and reactor coolant system are prevented from exceeding their safety limits. The various reactor trip circuits automatically open the reactor trip breakers whenever a parameter monitored by the RPS deviates from an allowed range. The RPS consists of four instrument channels for redundancy. The plant safety analyses are based on the trip setpoints given in Table 2.3-1 plus calibration and instrumentation errors.~~

Nuclear Overpower

~~A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.~~

~~During normal plant operation with all reactor coolant pumps operating, a reactor trip is initiated when the reactor power level reaches 105.5% of rated power. Adding to this the possible variation in trip setpoint due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis. (1)~~

A4

TABLE 3.3.1-1 ~~2.3-1~~ T3.3.1-1 Note a T3.3.1-1 Note b.

Reactor Protective System Trip Setting Limits

Func #	Function ~~RPS Trip~~	Allowable Value ~~RPS Trip Setpoint~~	Allowable Value ~~Shutdown~~ ~~Bypass~~
1. ~~1.~~	Nuclear Overpower	1.a 105.5% Rated Power	1.b 5.0% Rated Power(1) LA1
8. ~~2.~~	Flux/Flow/Imbalance	Axial Power Imbalance RPS Maximum Allowable Setpoints in the Core Operating Limits Report	~~Bypassed~~
7. ~~3.~~	Pump Monitors	At power operation >2.0% Rated Power and loss of two pumps	~~Bypassed~~
3+11 ~~4.~~	High Reactor Coolant System Pressure	3 2355 psig	11. 1720(1) LA1
4. ~~5.~~	Low Reactor Coolant System Pressure	1800 psig	~~Bypassed~~
5. ~~6.~~	Variable Low Reactor Coolant System Pressure	Variable Low RCS Pressure RPS Maximum Allowable Setpoints in the Core Operating Limits Report	~~Bypassed~~
2. ~~7.~~	High Reactor Coolant Temperature	618°F	618°F
6. ~~8.~~	High Reactor Building Pressure	4 psig	4 psig

~~(1) Administratively controlled reduction set only during reactor shutdown.~~

~~(2) Automatically set when other segments of the RPS are bypassed.~~

LA1

Add Function 9 + 10 Allowable Value - M9

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operation Safety Instrumentation

A1

Add LCO 3.3.1, T 3.3.1-1 Applicabilities for Funct. 1.6+11

Applicability

Applies to unit instrumentation and control systems.

M2

Objective

Add T 3.3.1-1 Notes c+d

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

Change Applic for Funct. 9+10 to ≥30% + 2% respectively L2

LCO + Applic.

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7. MODES 1+2 A1

ACT B

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D. A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light. LA8

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved. (SEE 3.3.10)

TSC 95-03

Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

A1 <Except as marked>

LA3 LCO RA C.1

TABLE ~~3.5.1-1~~ 3.3.1-1
~~INSTRUMENTS OPERATING CONDITIONS~~

<SEE 3.3.9 + 3.3.10>

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions of Column C Cannot Be Met
1. Nuclear Instrumentation Wide Range Channels	4	NA	2	Bring to hot shutdown within 12 hours (b)
2. Nuclear Instrumentation Source Range Channels	4	NA	2	Bring to hot shutdown within 12 hours (b) (c)
3. RPS Manual Pushbutton	1	1	1	Bring to hot shutdown within 12 hours <SEE 3.3.2>
1.a 1.b ~~4.~~ RPS Power Range Instrument Channels	4	2	3(a)	Bring to ~~hot shutdown~~ MODE 3 within 12 hours
2 ~~5.~~ RPS Reactor Coolant Temperature Instrument Channels	4	2	3(a)	Bring to ~~hot shutdown~~ MODE 3 within 12 hours
5 ~~6.~~ RPS Pressure-Temperature Instrument Channels	4	2	3(a)	Bring to ~~hot shutdown~~ MODE 3 within 12 hours
8 ~~7.~~ RPS Flux Imbalance Flow Instrument Channels	4	2	3(a)	Bring to ~~hot shutdown~~ MODE 3 within 12 hours
~~8.~~ RPS Reactor Coolant Pressure 3+11 ~~a.~~ High Reactor Coolant Pressure Instrument Channels	4	2	3(a)	Bring to ~~hot shutdown~~ MODE 3 within 12 hours
4 ~~b.~~ Low Reactor Coolant Pressure Channels	4	2	3(a)	Bring to ~~hot shutdown~~ MODE 3 within 12 hours
7 ~~9.~~ RPS Power-Number of Pumps Instrument Channels	4	2	3(a)	Bring to ~~hot shutdown~~ MODE 3 within 12 hours ~~(b)~~ LA4

LA3

M1 Add RA C.2 for ITS Funct. 1.a + 2-8

M30 Replace Column D Action with RA D.1 for ITS Functions 1.b + 11

A1 <Except as marked>

TABLE ~~3.5.1-1~~ 3.3.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

LCO

FUNCTIONAL ~~UNIT~~	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
6 ~~10~~. RPS High Reactor Building Pressure Channels	4	2	3(a)	RAC.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours
~~11~~. RPS Anticipatory Reactor Trip System				
9 ~~a~~. Loss of Turbine	4	2	3(a)	RAE.1 Reduce THERMAL POWER < 30% within 6 hrs. ~~Bring to hot shutdown within 12 hours~~
10 ~~b~~. Loss of Main Feedwater	4	2	3(a)	RAF.1 Reduce THERMAL POWER < 2% within 12 hours ~~Bring to hot shutdown within 12 hours~~
12. ESF High Pressure Injection System and Reactor Building Isolation (Non-essential Systems)				
a. Analog Reactor Coolant Pressure Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
c. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
d. Digital Logic Channels (1 and 2)	2	1	2	Bring to hot shutdown within 24 hours (e)

LA3

L2

<SEE 3.3.5, 6 + 7>

Specification 3.3.1

A 147/147/144
4/25/86

Page 5 of 10

Specification 3.3.1

A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

~~NOTES:~~

L1

with one channel inoperable

~~(a)~~ ACT A For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic. LA9

within one hour

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4 x 10^{-4} % rated power, hot shutdown is not required.

<SEE 3.3.9 + 10>

~~(d) (Deleted)~~

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours.

<SEE 3.3.5, 6, 7 + 15>

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state. LA4

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

4.1 OPERATIONAL SAFETY REVIEW

A1 <Except as marked>

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

~~4.1.1~~ SR Note The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1. <SEE 3.3.5, 3.3.6 + 3.3.7>

4.1.2 The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2. <SEE 3.1, 3.4, 3.5, 3.7>

4.1.3 Required sampling should be performed as detailed in Table 4.1-3. <SEE 3.4, 3.5, 3.7 + 3.9>

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in FSAR Chapter 16. <SEE 5.0>

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days. <SEE 3.2>

Bases

A2

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

A1 <Except as marked>

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

<SEE 3.3.3 + 4>

Function ~~Channel Description~~	Check	Test	Calibrate		Remarks
~~1. Protective Channel Coincidence Logic in the Reactor Trip Modules~~	~~NA~~	~~MO~~	~~NA~~		
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
1.a ~~3.~~ Power Range Amplifier SR 3.3.1.2	~~BS(1)~~ 24 hrs L9	NA	(1) For Funct. 8 only	(1)	Heat balance check ~~each shift~~ 24 hrs L9 Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
1.a 1.b 8 ~~4.~~ Power Range Nuclear Overpower Nuclear Overpower Flux/Flow Imbalance A5	SR 3.3.1.1 ~~BS~~ 12 hrs A9	SR 3.3.1.5 45 Days STB	SR 3.3.1.3/4 ~~MO~~(1)(2) 31 days A9 For Funct. 1.a + 8 only	(1) (2)	Using incore instrumentation. SR 3.3.1.3 Axial offset upper and lower chambers after each startup if not done previous week. L8
~~5. Wide Range~~	~~BS(1)~~	~~PS~~	~~NA~~	~~(1)~~	~~When in service.~~ <SEE 3.3.8, 9 + 10>
~~6. Source Range~~	~~BS(1)~~	~~PS~~	~~NA~~	~~(1)~~	~~When in service.~~ <See 3.3.8, 9, 10, 3.9.2>
2 ~~7.~~ Reactor Coolant Temperature	SR 3.3.1.1 ~~BS~~	SR 3.3.1.5 45 Days STB	SR 3.3.1.6 ~~RF~~		Add SR 3.3.1.2 Note L14
3 11 ~~8.~~ High Reactor Coolant Pressure	~~BS~~	45 Days STB	~~RF~~		Add SR 3.3.1.3 Note L15
4 ~~9.~~ Low Reactor Coolant Pressure	~~BS~~ 12 hrs A9	45 Days STB A9	~~RF~~ 18 months A9		Add SR 3.3.1.4 Note L4
8 ~~10.~~ Flux-Reactor Coolant Flow Comparator	~~BS~~	45 Days STB	~~RF~~		Add SR 3.3.1.6 Note for Functions 2, 3, 4, 5, 6, 7, 8, 9, 10 + 11 L20
5 ~~11.~~ Reactor Coolant Pressure Temperature Comparator	~~BS~~	45 Days STB	~~RF~~		Add SR 3.3.1.6 for Functions 1.a + 1.b M35

Amendment No. 223 (Unit 1)
Amendment No. 223 (Unit 2)
Amendment No. 220 (Unit 3)

Specification 3.3.1

A1 <Except as marked>

Table ~~4.1-1~~ 3.3.1-1 (CONTINUED)

Function ~~Channel Description~~	SR 3.3.1.1 Check	SR 3.3.1.5 Test	SR 3.3.1.6 Calibrate	Remarks
7. ~~12.~~ Pump-Flux Comparator	~~ES~~ 12 hours A9	45 Days STB	~~RF~~ 18 months A9	
6. 13. High Reactor Building Pressure M17	~~DA~~ 12 hours	45 Days STB	~~RF~~ 18 months	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	Includes Reactor Building Isolation of non-essential systems
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	NA	MO	NA	
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	Reactor Building isolation includes essential systems
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

<SEE 3.3.5 + 7>

3.3.1-1 A1

Table ~~4.1-1~~ (CONTINUED)

Function ~~Channel Description~~		Check	Test	Calibrate	Remarks
~~49~~	Emergency Feedwater Flow Indicators	MO	NA	RF	<SEE 3.3.8>
50.	PORV and Safety Valve Position Indicators	MO	NA	RF	R1
9. ~~51.~~	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	SR 3.3.1.5 45 Days ~~STB~~ A9	SR 3.3.1.6 ~~RF~~ 18 months A9	
10. ~~52.~~	RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
	a) Control Oil Pressure Switches	NA	45 Days ~~STB~~	~~RF~~ 18 months A9	
53.	Emergency Feedwater Initiation Circuits				
	a) Control Oil Pressure Switches	NA	MO	RF	<SEE 3.3.14>
54.	Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	RF	TMI Item II F.1.3 <SEE 3.3.8>

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Specification 3.3.1

Specification 3.3.2

M3

Add LCO Applic. of MODES 3, 4 + 5 with any CRD trip breaker in the closed position + the CRD System capable of rod withdrawal

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operation Safety Instrumentation

A1

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

MODE 1 + 2 - M3

LCO + Applic.

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met ~~with the exception of items 20, 21, and 22. For items 20, 21, and 22, the requirements are specified in Specification 3.5.7.~~ A1

L3

Add ACT A

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D. A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light. (SEE 3.3.1)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved. (SEE 3.3.10)

A1 <Except as marked>

3.3.1-1

TABLE ~~3.3.1-1~~

~~INSTRUMENTS OPERATING CONDITIONS~~

LA3

~~FUNCTIONAL UNIT~~	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions of Column C Cannot Be Met
1. Nuclear Instrumentation Wide Range Channels	4	NA	2	Bring to hot shutdown within 12 hours (b)
2. Nuclear Instrumentation Source Range Channels	4	NA	2	Bring to hot shutdown within 12 hours (b) (c)
~~3.~~ RPS Manual Pushbutton (LA3)	V	1	LCO 1	RAB.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours
M1 — Add RAB.2				
4. RPS Power Range Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
5. RPS Reactor Coolant Temperature Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
6. RPS Pressure-Temperature Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
7. RPS Flux Imbalance Flow Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
8. RPS Reactor Coolant Pressure a. High Reactor Coolant Pressure Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Low Reactor Coolant Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
9. RPS Power-Number of Pumps Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours (h)

<SEE 3.3.9 + 10>

<SEE 3.3.1>

Add ACTION C — M3

Specification 3.3.2

(A1)

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

SR 3.3.2.1

(SEE 3.3.5, 6+7)

4.1.2 The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2. (SEE 3.1, 3.4, 3.5 + 3.7)

4.1.3 Required sampling should be performed as detailed in Table 4.1-3. (SEE 3.4, 3.5, 3.7 + 3.9)

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in FSAR Chapter 16. (SEE 5.0)

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days. (SEE 3.2)

Bases

(A2)

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

A1 <except as marked>

~~Table 4.1-1 (CONTINUED)~~

~~Channel Des~~cription	Check	SR 3.3.2.1 Test	~~Calibrate~~	~~Remarks~~
30. Borated Water Storage Tank Level Indicator	WE	NA	RF	
31. Boric Acid Mix Tank:				
a. Level	NA	NA	AN	<SEE 3.3.8>
b. Temperature	MO	NA	AN	
32. Concentrated Boric Acid Storage Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
33. Containment Temperature	NA	NA	RF	<SEE 3.2>
34. Incore Neutron Detectors	MO(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout.
35. Emergency Plant Radiation Instruments	MO(1)	NA	RF	(1) Battery check. <SEE 3.3.8>
36. Environmental Monitors	MO(1)	NA	RF	(1) Check functioning.
37. Reactor Manual Trip	NA	PS	NA	SR 3.3.2.1; SR 3.3.2.1 Frequency A9
38. Reactor Building Emergency Sump Level	NA	NA	RF	
39. Steam Generator Water Level	WE	NA	RF	<SEE 3.3.8>
40. Turbine Overspeed Trip	NA	NA	RF	

4.1-6

12/11/87
Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

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Specification 3.3.2

Specification 3.3.3

M3

Add LCO Applic of MODES 3, 4 + 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operation Safety Instrumentation

A1

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

MODE 1 + 2 — M3

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of items 20, 21, and 22. For items 20, 21, and 22, the requirements are specified in Specification 3.5.7. A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D. A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light. (SEE 3.3.1)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved. (SEE 3.3.10)

A1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
15. ESF Reactor Building Spray System				
a. Analog Reactor Building High Pressure Instrument Channel	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
c. Digital Logic Channels (7 and 8)	2	1	2	Bring to hot shutdown within 24 hours (e)
16. Turbine Stop Valves Closure	2	1	2	Bring to hot shutdown within 24 hours (e)
LCO ~~17.~~ Protective Channel Coincidence Logic in the Reactor Trip Modules	4 logic channels; A, B, C, and D	AB or AD or BC or CD	4 RAA.1.1 RAA.1.2	See Note (f)
18. CRD Breakers	1 AC Breaker and 2 DC Breakers per trip system		1 AC Breaker and 2 DC Breakers per trip system	See Note (i)
19. SCR Control Relays E and F	4 SCR Control Relays per trip system		4 SCR Control Relays per trip system	See Note (j)

LA3

<SEE 3.3.5, 6+7>

<SEE 3.3.15>

<SEE 3.3.4>

A 148,148,145
8/20/86

Page 27 5

Specification 3.3.3

A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4 x 10^{-4} % rated power, hot shutdown is not required.

<SEE 3.3.9 + 10>

~~(d) (Deleted)~~

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours.

<SEE 3.3.5, 6, 7 + 15>

RA A1.1 (f) 1. ~~Place the inoperable Reactor Trip Module output in the tripped condition~~ within one hour or Trip the associated trip breaker A3

RA A.1.2 2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

~~(g) (Deleted)~~ Add RA A.2 M4

(h) The RCP monitors provide inputs to this logic. For operability to be met ~~either~~ all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state. <SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

Add ACTION B M5

Add ACTION C M3

Specification 3.3.3.

A1

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

SR 3.3.3.1

(SEE 3.3.5, 6+7)

4.1.2 ~~The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2.~~ (SEE 3.1, 3.4, 3.5+3.7)

4.1.3 ~~Required sampling should be performed as detailed in Table 4.1-3.~~ (SEE 3.4, 3.5, 3.7+3.9)

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in FSAR Chapter 16. (SEE 5.0)

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days. (SEE 3.2)

Bases

A2

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

Table 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

A1

	Channel Description	Check	Test	Calibrate		Remarks
1.	Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	SR 3.3.3.1 MO 31 days	NA		
2.	Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3.	Power Range Amplifier	BS(1)	NA	(1)	(1)	Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4.	Power Range	BS	45 Days STB	MO(1)(2)	(1) (2)	Using incore instrumentation. Axial offset upper and lower chambers after each startup if not done previous week.
5.	Wide Range	BS(1)	PS	NA	(1)	When in service.
6.	Source Range	BS(1)	PS	NA	(1)	When in service.
7.	Reactor Coolant Temperature	BS	45 Days STB	RF		
8.	High Reactor Coolant Pressure	BS	45 Days STB	RF		
9.	Low Reactor Coolant Pressure	BS	45 Days STB	RF		
10.	Flux-Reactor Coolant Flow Comparator	BS	45 Days STB	RF		
11.	Reactor Coolant Pressure Temperature Comparator	BS	45 Days STB	RF		

A9

<SEE 3.3.4>

<SEE 3.3.8, 9, 10 + 3.9.2>

<SEE 3.3.1>

Specification 3.3.3

Specification 3.3.4

(M3)

Add LCO Applic of MODES 3, 4 + 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal

3.5 INSTRUMENTATION SYSTEMS

(A1)

3.5.1 ~~Operation Safety Instrumentation~~

~~Applicability~~

~~Applies to unit instrumentation and control systems.~~

~~Objective~~

~~To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.~~

~~Specifications~~

MODE 1 + 2 (M3)

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met ~~with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7.~~ (A1)

3.5.1.2 ~~In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D.~~ (A1)

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light. <SEE 3.3.1>

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved. <SEE 3.3.10>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

(A1)

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
15. ESF Reactor Building Spray System				<SEE 3.3.5, 6 + 7>
a. Analog Reactor Building High Pressure Instrument Channel	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
c. Digital Logic Channels (7 and 8)	2	1	2	Bring to hot shutdown within 24 hours (e)
16. Turbine Stop Valves Closure	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.15>
17. Protective Channel Coincidence Logic in the Reactor Trip Modules	4 logic channels; A, B, C, and D	AB or AD or BC or CD	4	See Note (f) <SEE 3.3.3>
LCO 18. CRD Breakers	~~1 AC Breaker and 2 DC Breakers per trip system~~ (LA3)		LCO ~~1~~ 2 AC Breaker and 2 DC Breakers ~~per trip system~~ pair	ACTIONS A+B See Note (i)
LCO 19. ~~SCR~~ ETA Control Relays ~~E and F~~ (LA10)	4 SCR Control Relays per trip system		LCO ~~4~~ 8 ~~SCR~~ ETA Control Relays ~~per trip system~~ (A21)	ACTION C See Note (j)

3.5-5b

A 148,148,145
8/20/86

Page 2 of 6

Specification 3.3.4

Specification 3.3.4

A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4 x 10^{-4} % rated power, hot shutdown is not required.

<SEE 3.3.9 + 10>

~~(d) (Deleted)~~

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours.

<SEE 3.3.5, 6, 7 + 15>

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met ~~either all~~ RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state. <SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or trip the CRD trip breaker RA B.1 L10

RA B.2

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours, or place the breaker in trip ~~in the next hour~~ M10

RA A.1

RA A.2 or remove power from the CRD trip breaker

Add ACTIONS Note A6

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Amendment No. _______ (Unit 1)
Amendment No. _______ (Unit 2)
Amendment No. _______ (Unit 3)

Specification 3.3.4

A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

or more — M11

(j) 1. RAC.2 With one SCR Control Relay inoperable in logic channel C or D, restore the inoperable SCR Control Relay to OPERABLE status in ~~48 hours~~ or remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within the next hour. 1 hour — M11

one — M11

2. RAC.2 With ~~two~~ or more SCR Control Relays inoperable in logic channel C or D, remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within one hour. A32

(k) Requirement of 3 channels can be met with one of three channels placed in trip. The affected channel shall be placed in trip within 4 hours of discovery. <SEE 3.3.11>

(l) 1 of 2 digital channels or manual pushbutton can be disabled for up to 72 hours and still meet the requirements of this column.

<SEE 3.3.12 + 13>

Add RAC.1 — L11

Add ACTION D — M5

Add ACTION E — M3

Specification 3.3.4

A1

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

SR 3.3.4.1

(SEE 3.3.5, 6+7)

4.1.2 ~~The frequency~~ and type of surveillance required for selected equipment shall be as stated in Table 4.1-2. (SEE 3.1, 3.4, 3.5+3.7)

4.1.3 Required sampling should be performed as detailed in Table 4.1-3. (SEE 3.4, 3.5, 3.7+3.9)

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in FSAR Chapter 16. (SEE 5.0)

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days. (SEE 3.2)

Bases

A2

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

A1

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate		Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA		
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	SR 3.3.4.1 MO(1) 31 days	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	BS(1)	NA	(1)	(1)	Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	BS	45 Days STB	MO(1)(2)	(1) (2)	Using incore instrumentation. Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	BS(1)	PS	NA	(1)	When in service.
6. Source Range	BS(1)	PS	NA	(1)	When in service.
7. Reactor Coolant Temperature	BS	45 Days STB	RF		
8. High Reactor Coolant Pressure	BS	45 Days STB	RF		
9. Low Reactor Coolant Pressure	BS	45 Days STB	RF		
10. Flux-Reactor Coolant Flow Comparator	BS	45 Days STB	RF		
11. Reactor Coolant Pressure Temperature Comparator	BS	45 Days STB	RF		

LA10 A9 LA2 LA2

<SEE 3.3.3>

<SEE 3.3.8, 9, 10> + 3.9.2

<SEE 3.3.1>

Specification 3.3.4

Amendment No. 223 (Unit 1)
Amendment No. 223 (Unit 2)
Amendment No. 220 (Unit 3)

A1 <Except as marked>

3.5.3 Engineered Safety Features Protective System Actuation Setpoints

Applicability

This specification applies to the engineered safety features protective system actuation setpoints.

Objective

To provide for automatic initiation of the engineered safety features protective system in the event of a breach of RCS integrity.

Specification

The engineered safety features protective actuation setpoints and permissible bypasses shall be as follows:

LA6

Functional Unit Parameter		Action	Setpoint Table 3.3.5-1 AV
High Reactor Building Pressure-High High	4	~~Reactor Building Spray~~	≤15 psig
Reactor Building Pressure-High	3	High-Pressure Injection	≤4 psig
	3	Low-Pressure Injection	≤4 psig
	3	Start Reactor Building Cooling & Reactor Building Isolation (Essential and Non-essential Systems)	≤4 psig
	3	Penetration Room Ventilation	≤4 psig
~~Lower~~ Reactor Coolant System Pressure -Low	1	High Pressure Injection (1) & Reactor Building Isolation (Non-essential systems)	≥1500 psig
T 3.3.5-1 RCS Press.-Low Low	2	Low Pressure Injection (2)	≥500 psig

Para. 1 Applic (1) May be bypassed below 1750 psig and is automatically reinstated above 1750 psig.

Para. 2 Applic (2) May be bypassed below 900 psig and is automatically reinstated above 900 psig.

A3

M6

MODES 1, 2, 3 + 4 for RB press parameters

3.5 INSTRUMENTATION SYSTEMS

A1

3.5.1 Operation Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

≥1750 psig for RCS press-Low, ≥900 psig for RCS press-Low, Low

A3

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, ~~with the exception of items 20, 21, and 22.~~ For items 20, 21, and 22, the requirements are specified in Specification 3.5.7.

A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D.

A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light.

(SEE 3.3.1)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved.

(SEE 3.3.10)

Add ACTION A — L5

Add ACTIONS Note — A6

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Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

A1

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
10. RPS High Reactor Building Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
11. RPS Anticipatory Reactor Trip System				
a. Loss of Turbine	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Loss of Main Feedwater	4	2	3(a)	Bring to hot shutdown within 12 hours
				<SEE 3.3.1>
12. ESF High Pressure Injection System and Reactor Building Isolation (Non-essential Systems)				
LCO — Parameter 1 a. Analog Reactor Coolant Pressure Instrument Channels	3	2 LA3	LCO 3	RA B.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours (e) A1
Parameter 3 b. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	RA B.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours (e) A1
				<SEE 3.3.6>
c. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
d. Digital Logic Channels (1 and 2)	2	1	2	Bring to hot shutdown within 24 hours (e)
				<SEE 3.3.7>

A 147/147/144
4/25/86

Specification 3.3.5

A1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
13. ESF Low Pressure Injection System				
LCO Parameter 2 ~~a.~~ Analog Reactor Coolant Pressure Instrument Channels	3	2 LA3	LCO 3	RA B.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours (e)
Parameter 3 ~~b.~~ Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	RA B.1 Bring to ~~hot shutdown~~ A1 MODE 3 within 12 hours (e)
c. Digital Logic Manual Pushbutton	2	1	2	<SEE 3.3.6> Bring to hot shutdown within 12 hours. (e)
d. Digital Logic Channels (3 and 4)	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.7>
14. ESF Reactor Building Isolation (Essential Systems) & Reactor Building Cooling System				
LCO Parameter 3 a. Analog Reactor Building 4 PSIG Instrument Channels	3	2 LA3	LCO 3	RA B.1 Bring to ~~hot shutdown~~ A1 MODE 3 within 12 hours (e)
b. Digital Logic Manual Pushbutton	2	1	2	<SEE 3.3.6> Bring to hot shutdown within 12 hours (e)
c. Digital Logic Channels (5 and 6)	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.7>

3.5-5a

A 117/117/114
11/22/82

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Specification 3.3.5

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

A1

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
15. ESF Reactor Building Spray System				
LCO a. Analog Reactor Building High Pressure Instrument Channel (Parameter 4)	~~3~~	~~2~~ LA3	LCO [3	RA B.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours (e)
b. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e) <SEE 3.3.6>
c. Digital Logic Channels (7 and 8)	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.7>
16. Turbine Stop Valves Closure	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.15>
17. Protective Channel Coincidence Logic in the Reactor Trip Modules	4 logic channels; A, B, C, and D	AB or AD or BC or CD	4	See Note (f) <SEE 3.3.3>
18. CRD Breakers	1 AC Breaker and 2 DC Breakers per trip system		1 AC Breaker and 2 DC Breakers per trip system	See Note (i)
19. SCR Control Relays E and F	4 SCR Control Relays per trip system		4 SCR Control Relays per trip system	See Note (j)

<SEE 3.3.4>

A 148,148,145
8/20/86

Specification 3.3.5

A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4×10^{-4} % rated power, hot shutdown is not required.

<SEE 3.3.9 + 10>

~~(d) (Deleted)~~

RA B.2.3 for Parameters 3 + 4 + Note

(e) If minimum conditions are not met ~~within 48 hours after hot shutdown~~ the unit shall be in ~~cold shutdown~~ MODE 5 within ~~24~~ hours. 36 M7

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state.

<SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

Add RA B.2.1 + Note for Parameter 1
Add RA B.2.2 + Note for Parameter 2

A8

Specification 3.3.5

A1 <Except as marked>

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

<SEE 3.3.1-4>

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

SRs

4.1.2 The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2. <SEE 3.1, 3.4, 3.5 + 3.7>

4.1.3 Required sampling should be performed as detailed in Table 4.1-3. <SEE 3.4, 3.5, 3.7 + 3.9>

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in FSAR Chapter 16. <SEE 5.0>

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days. <SEE 3.2>

Bases

A2

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

A1 <Except as marked

Table 4.1-1 (CONTINUED)

Channel Description	Check	Test	Calibrate	Remarks
12. Pump-Flux Comparator	ES	45 Days STB	RF	<SEE 3.3.1>
13. High Reactor Building Pressure	DA	45 Days STB	RF	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	NA	MO	NA	<SEE 3.3.7> Includes Reactor Building Isolation of non-essential systems LA6
15. High Pressure Injection Analog Channels:				
Parameter	SR 3.3.5.1	SR 3.3.5.2	SR 3.3.5.3	Add SR 3.3.5.2 Note L6
1 ~~a.~~ Reactor Coolant Pressure	~~ES~~ 12 hrs	~~MO~~ 31 days	~~RF~~ 18 months	A9
3 ~~b.~~ Reactor Building Pressure (4 psig)	~~ES~~	~~MO~~	~~RF~~	
16. Low Pressure Injection Logic	NA	MO	NA	<SEE 3.3.7>
17. Low Pressure Injection Analog Channels:				A9
2 ~~a.~~ Reactor Coolant Pressure	~~ES~~ 12 hrs	~~MO~~ 31 days	~~RF~~ 18 months	
3 ~~b.~~ Reactor Building Pressure (4 psig)	~~ES~~	~~MO~~	~~RF~~	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	NA	MO	NA	<SEE 3.3.7> Reactor Building isolation includes essential systems LA6
~~19.~~ 3 Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	~~ES~~ 12 hrs	~~MO~~ 31 days	~~RF~~ 18 months	A9

A1 <Except as marked>

TABLE 4.1-1 (Continued)

Channel Description	Check	Test	Calibrate	Remarks
20. Reactor Building Spray System Logic	NA	MO	NA	<SEE 3.3.7>
Parameter 4 21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	~~NA~~ Add SR 3.3.5.1 M21	SR3.3.5.2 ~~MO~~ 31 days A9	SR 3.3.5.3 ~~RF~~ 18 months	
22. Pressurizer Temperature	ES	NA	RF	R1
23. Control Rod Absolute Position	ES(1)	NA	RF(2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.
24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel. <SEE 3.1>
25. Core Flood Tanks				
a. Pressure	ES	NA	RF	R1
b. Level	ES	NA	RF	
26. Pressurizer Level	ES	NA	RF	<SEE 3.3.8>
27. Letdown Storage Tank Level	DA	NA	RF	R1
~~28. Delete~~				
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	RF	<SEE 3.3.8>

A 125/125/122
1/16/84

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Specification 3.3.5

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operation Safety Instrumentation

A1

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety

M8

Specifications

MODES 1+2
MODES 3+4 when associated ES equipment is required to be OPERABLE

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7. A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D. A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light. SEE 3.3.1

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved. SEE 3.3.10

Add ACTIONS Note A6

Add ACTION A L7

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Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

A1

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
10. RPS High Reactor Building Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
11. RPS Anticipatory Reactor Trip System				
a. Loss of Turbine	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Loss of Main Feedwater	4	2	3(a)	Bring to hot shutdown within 12 hours
~~12.~~ ESF High Pressure Injection System and Reactor Building Isolation (Non-essential Systems)				
a. Analog Reactor Coolant Pressure Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
LCO 3.3.6.a c. Digital Logic Manual Pushbutton	2	1	2	RA B.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours (e) A1
d. Digital Logic Channels (1 and 2)	2	1	2	Bring to hot shutdown within 24 hours (e)

<SEE 3.3.1>

Keowee Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input

A33

<SEE 3.3.5>

LA3

<SEE 3.3.7>

3.5-5

A 147/147/144
4/25/86

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Specification 3.3.6

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

A1

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) ~~Operator~~ Action If Conditions Of Column C Cannot Be Met
13. ESF Low Pressure Injection System and Reactor Building Essential Isolation A34				<SEE 3.3.5>
a. Analog Reactor Coolant Pressure Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
LCO 3.3.6.b c. Digital Logic Manual Pushbutton	~~2~~ LA3	~~1~~	2	RA B.1 Bring to ~~hot shutdown~~ MODE 3 A1 within 12 hours. (e)
d. Digital Logic Channels (3 and 4)	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.7>
14. ESF Reactor Building Isolation (Essential Systems) & Reactor Building Cooling System and Penetration Room Ventilation A35				<SEE 3.3.5>
a. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
LCO 3.3.6.c b. Digital Logic Manual Pushbutton	~~2~~ LA3	~~1~~	2	RA B.1 Bring to ~~hot shutdown~~ MODE 3 A1 within 12 hours (e)
c. Digital Logic Channels (5 and 6)	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.7>

3.5-5a

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A1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

FUNCTIONAL UNIT	(A) TOTAL NO OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
~~15.~~ ESF Reactor Building Spray System				
a. Analog Reactor Building High Pressure Instrument Channel	3	2	3	Bring to hot shutdown within 12 hours (e) <SEE 3.3.5>
LCO 3.3.6.d b. Digital Logic Manual Pushbutton	2	1 LA3	2	RA B.1 Bring to ~~hot shutdown~~ MODE 3 within 12 hours (e)
c. Digital Logic Channels (7 and 8)	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.7>
16. Turbine Stop Valves Closure	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.15>
17. Protective Channel Coincidence Logic in the Reactor Trip Modules	4 logic channels; A, B, C, and D	AB or AD or BC or CD	4	See Note (f) <SEE 3.3.3>
18. CRD Breakers	1 AC Breaker and 2 DC Breakers per trip system		1 AC Breaker and 2 DC Breakers per trip system	See Note (i)
19. SCR Control Relays E and F	4 SCR Control Relays per trip system		4 SCR Control Relays per trip system	See Note (j)

<SEE 3.3.4>

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A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4×10^{-4}% rated power, hot shutdown is not required.

<SEE 3.3.9 + 10>

~~(d) (Deleted)~~

RA B.2 (e) If minimum conditions are not met ~~within 48 hours after hot shutdown~~ the unit shall be in ~~cold shutdown~~ MODE 5 within ~~24~~ 36 hours. MT

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state. <SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

A1 <Except as marked>

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

<SEE 3.3.1-4>

4.1.1 The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

SRs

4.1.2 The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2. <SEE 3.1, 3.4, 3.5 + 3.7>

4.1.3 Required sampling should be performed as detailed in Table 4.1-3. <SEE 3.4, 3.5, 3.7 + 3.9>

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in FSAR Chapter 16. <SEE 5.0>

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days. <SEE 3.2>

Bases

A2

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

A1 <except as marked>

Table 4.1-1 (CONTINUED)

LA6

Channel Description	Check	SR 3.3.6.1 ~~Test~~	Calibrate	Remarks
41. Engineered Safeguards Channel 1 HP Injection & Reactor Building Isolation Manual Trip	NA	~~RF~~ 18 months	NA	Includes Reactor Building isolation of non-essential systems only
42. Engineered Safeguards Channel 2 HP Injection & Reactor Building Isolation Manual Trip	NA	~~RF~~	NA	Includes Reactor Building isolation of non-essential systems only
43. Engineered Safeguards Channel 3 LP Injection Manual Trip	NA	~~RF~~	NA	
44. Engineered Safeguards Channel 4 LP Injection Manual Trip	NA	~~RF~~	NA	
45. Engineered Safeguards Channel 5 RB Isolation & Cooling Manual Trip	NA	~~RF~~	NA	Includes Reactor Building isolation of essential systems only
46. Engineered Safeguards Channel 6 RB Isolation & Cooling Manual Trip	NA	~~RF~~	NA	Includes Reactor Building isolation of essential systems only
47. Engineered Safeguards Channel 7 Spray Manual Trip	NA	~~RF~~	NA	
48. Engineered Safeguards Channel 8 Spray Manual Trip	NA	~~RF~~	NA	

A9

A 92/92/89
1/28/81

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Specification 3.3.6

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operation Safety Instrumentation

A1

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

M8

MODES 1 + 2
MODES 3 + 4 when associated ES equipment is required to be OPERABLE

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7.

A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D.

A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light.

(SEE 3.3.1)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved.

(SEE 3.3.10)

Add ACTIONS Note — A6

A1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
10. RPS High Reactor Building Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
11. RPS Anticipatory Reactor Trip System				
a. Loss of Turbine	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Loss of Main Feedwater	4	2	3(a)	Bring to hot shutdown within 12 hours
				<SEE 3.3.1>
12. ESF High Pressure Injection System and Reactor Building Isolation (Non-essential Systems)				
				<SEE 3.3.5>
a. Analog Reactor Coolant Pressure Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
				<SEE 3.3.6>
c. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
LCO d. Digital Logic Channels (1 and 2)	2	1 LA3	2	~~Bring to hot shutdown within 24 hours (e)~~ L12

RA A.1+A.2

Place associated components (s) in ESF configuration in 1 hour.
OR
Declare the associated component(s) inoperable in 1 hour.

3.5-5

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Specification 3.3.7

A1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
13. ESF Low Pressure Injection System				
a. Analog Reactor Coolant Pressure Instrument Channels	3	2	3	<SEE 3.3.5> Bring to hot shutdown within 12 hours (e)
b. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	Bring to hot shutdown within 12 hours (e)
c. Digital Logic Manual Pushbutton	2	1	2	<SEE 3.3.6> Bring to hot shutdown within 12 hours. (e)
LCO d. Digital Logic Channels (3 and 4)	2 LA3	1	2	~~Bring to hot shutdown within 24 hours (e)~~ L12 Replace with RA A.1 + A.2
14. ESF Reactor Building Isolation (Essential Systems) & Reactor Building Cooling System				
a. Analog Reactor Building 4 PSIG Instrument Channels	3	2	3	<SEE 3.3.5> Bring to hot shutdown within 12 hours (e)
b. Digital Logic Manual Pushbutton	2	1	2	<SEE 3.3.6> Bring to hot shutdown within 12 hours (e)
LCO c. Digital Logic Channels (5 and 6)	2 LA3	1	2	~~Bring to hot shutdown within 24 hours (e)~~

3.5-5a

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Specification 3.3.7

A1

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
15. ESF Reactor Building Spray System				
a. Analog Reactor Building High Pressure Instrument Channel	3	2	3	Bring to hot shutdown within 12 hours (e) <SEE 3.3.5>
b. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e) <SEE 3.3.6>
LCO c. Digital Logic Channels (7 and 8)	2	1 LA3	2 L12 Replace with RA A1+A2	Bring to hot shutdown within 24 hours (e)
16. Turbine Stop Valves Closure	2	1	2	Bring to hot shutdown within 24 hours (e) <SEE 3.3.15>
17. Protective Channel Coincidence Logic in the Reactor Trip Modules	4 logic channels; A, B, C, and D	AB or AD or BC or CD	4	See Note (f) <SEE 3.3.3>
18. CRD Breakers	1 AC Breaker and 2 DC Breakers per trip system		1 AC Breaker and 2 DC Breakers per trip system	See Note (i)
19. SCR Control Relays E and F	4 SCR Control Relays per trip system		4 SCR Control Relays per trip system	See Note (j) <SEE 3.3.4>

3.5-5b

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Specification 3.3.7

Specification 3.3.7

A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4 x 10^{-4} % rated power, hot shutdown is not required.

<SEE 3.3.9 + 10>

~~(d) (Deleted)~~

Add RA A.1 + A.2

L12

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours.

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state. <SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

Specification 3.3.7

A1 <Except as marked>

4.1 OPERATIONAL SAFETY REVIEW

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

<SEE 3.3.1-4>

4.1.1 SRs The frequency and type of surveillance required for Reactor Protective System and Engineered Safety Feature Protective System instrumentation shall be as stated in Table 4.1-1.

4.1.2 The frequency and type of surveillance required for selected equipment shall be as stated in Table 4.1-2. <SEE 3.1, 3.4, 3.5 + 3.7>

4.1.3 Required sampling should be performed as detailed in Table 4.1-3. <SEE 3.4, 3.5, 3.7 + 3.9>

4.1.4 The frequency and type of surveillance required for radioactive effluent monitoring instrumentation shall be as stated in FSAR Chapter 16. <SEE 5.0>

4.1.5 Using the Incore Instrumentation System, a power map shall be made to verify expected power distribution at periodic intervals not to exceed ten effective full power days. <SEE 3.2>

Bases

A2

Failures such as blown instrument fuses, defective indicators, and faulted amplifiers are, in many cases, revealed by alarm or annunciator action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear systems, when the unit is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation.

Calibration is performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers are calibrated (during steady-state operating conditions) when indicated neutron power exceeds core thermal power by more than two percent. During non-steady-state operation, the nuclear flux channels amplifiers are calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. Calibration checks are also performed following significant changes in core conditions (power level and control rod positions) in order to assure that the core thermal power indication during non-steady-state operations does not exceed the indicated neutron power by more than the tolerance (4% FP) assumed in the safety analysis for significant duration (e.g., 4 hours).

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system

A1

Table ~~4.1-1 (CONTINUED)~~ 3.3.7.1

~~Channel Description~~	~~Check~~	~~Test~~	~~Calibrate~~	~~Remarks~~
12. Pump-Flux Comparator	ES	45 Days STB	RF	<SEE 3.3.1>
13. High Reactor Building Pressure	DA	45 Days STB	RF	
14. High Pressure Injection & Reactor Building Isolation Logic (Non-essential systems)	~~NA~~ SR 3.3.7.1	~~MO~~ 31 days A9	~~NA~~	~~Includes Reactor Building Isolation of non-essential systems~~ LA6
15. High Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
16. Low Pressure Injection Logic	~~NA~~ SR 3.3.7.1	~~MO~~ 31 days A9	~~NA~~	<SEE 3.3.5>
17. Low Pressure Injection Analog Channels:				
a. Reactor Coolant Pressure	ES	MO	RF	
b. Reactor Building Pressure (4 psig)	ES	MO	RF	
18. Reactor Building Emergency Cooling and Isolation System Logic (Essential Systems)	~~NA~~ SR 3.3.7.1	~~MO~~ 31 days A9	~~NA~~	~~Reactor Building isolation includes essential systems~~ LA6
19. Reactor Building Emergency Cooling and Isolation System Analog Channel Reactor Building Pressure (4 psig)	ES	MO	RF	

TABLE 4.1-1 (Continued)

SR 3.3.7.1

A1

Channel Description	Check	~~Test~~	Calibrate	Remarks
20. Reactor Building Spray System Logic	~~NA~~	~~MO~~ 31 days	~~NA~~	A9
21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	RF	<SEE 3.3.5>
22. Pressurizer Temperature	ES	NA	RF	<SEE 3.3.5>
23. Control Rod Absolute Position	ES(1)	NA	RF(2)	(1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.
24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel. <SEE 3.1>
25. Core Flood Tanks				
a. Pressure	ES	NA	RF	<SEE 3.3.5>
b. Level	ES	NA	RF	<SEE 3.3.5>
26. Pressurizer Level	ES	NA	RF	<SEE 3.3.8>
27. Letdown Storage Tank Level	DA	NA	RF	<SEE 3.3.5>
28. ~~Delete~~				
29. High and Low Pressure Injection Systems Flow Channels	NA	NA	RF	<SEE 3.3.8>

Specification 3.3.7

Specification 3.3.8 for Function 14

A28 Add ACTIONS Note 1 for Funct. 14

Add ACTIONS Note 2 for Funct. 14 A17

Add ACTION G for Function 14 A20

<SEE 3.5.1>

3.3.3 Core Flood Tank (CFT) System

When the RCS is in a condition with pressure above 800 psig both CFT's shall be operable with the electrically operated discharge valves open and breakers locked open and tagged; a minimum level of 13 ± .44 feet (1040 ± 30 ft.[3]) and R1 ~~one level instrument channel per CFT,~~ a minimum boron concentration within the limit specified in the Core Operating Limits Report in each CFT; and pressure at 600 ± 25 psig ~~with one pressure instrument channel per CFT.~~ R1

3.3.4 Borated Water Storage Tank (BWST) A14

MODES 1, 2 + 3 A14

Applic. ~~When the RCS, with fuel in the core,~~ is in ~~a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F:~~

LCO T3.3.8-1 Function 14 a. The BWST shall have operable two level instrument channels.

A1

~~(1) Tests or maintenance shall be allowed on one channel of BWST level instrumentation provided the other channel is operable.~~

MODE 3 A1

RA E.1 (2) If the BWST level instrumentation is not ~~restored to meet the~~ requirements of Specification 3.3.4.a above within 24 hours, the

RA H.1 reactor shall be placed in ~~a hot shutdown condition~~ within 12 hours. If the requirements of Specification 3.3.4.a are not met ~~within 24 hours following hot shutdown,~~ the reactor shall be

RA H.2 placed in a condition with ~~RCS pressure below 350 psig and RCS temperature below 250°F within an additional~~ 24 hours. A14

M24 MODE 4 18 M24

b. The BWST shall contain a minimum level of 46 feet of water having a minimum concentration of boron within the limit specified in the Core Operating Limits Report at a minimum temperature of 50°F. The manual valve, LP-28, on the discharge line shall be locked open. If these requirements are not met, the BWST shall be considered unavailable and action initiated in accordance with Specification 3.2.

<SEE 3.5.4>

3.3.5 Reactor Building Cooling (RBC) System

a. When the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical:

(1) Two independent RBC trains, each comprised of an RBC fan, associated cooling unit, and associated ESF valves shall be operable. Valve LPSW-108 shall be locked open.

(2) Tests or maintenance shall be allowed on any component of the RBC system provided one train of the RBC and one train of the RBS are operable. If the RBC system is not restored to meet the requirements of Specification 3.3.5.a(1) above within 24 hours, the reactor shall be placed in a condition with RCS pressure below 350 psig and RCS temperature below 250°F within an additional 24 hours.

<SEE 3.6>

Specification 3.3.8
for Function 21

A1 <Except as marked>

3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

Specification

A11

MODES 1, 2 + 3

Applic. for Func. 21 3.4.1 The reactor shall not be heated above 250°F unless the following conditions are met:

a. Three emergency feedwater pumps (one steam driven pump capable of being driven from an operable steam supply system and two motor driven pumps) and associated manual initiation circuitry shall be operable. <SEE 3.3.14>

<SEE 3.7>

LCO T3.3.8-1 Func. 21 b. Two 100% emergency feedwater flow paths shall be operable. Each flow path shall have at least ~~one~~ flow indicator operable. two — M36

3.4.2 In addition to the requirements of 3.4.1, prior to criticality, the automatic initiation circuitry associated with loss of main feedwater pumps as sensed by low hydraulic oil pressure shall be operable.

3.4.3 During operation greater than 250°F, the provisions of 3.4.1 and 3.4.2 may be modified to permit the following conditions:

a. One motor driven emergency feedwater pump may be inoperable for a period of up to seven days. If the inoperable pump is not restored to operable status within 7 days, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

7 days — L37

RA C.1 b. ~~One turbine driven emergency feedwater pump or one emergency feedwater flow path~~ may be inoperable for a period of up to ~~72~~ hours. If the inoperable pump or flow

RA H.1 path is not restored to operable status within 72 hours the unit will be ~~at hot shutdown~~ MODE 3

RA H.2 within an additional 12 hours and below 250°F in another ~~12~~ 6 hours. MODE 4 — M37

c. Two motor driven emergency feedwater pumps may be inoperable for a period of up to 12 hours. If at least one pump is not restored to operable status within 12 hours, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

<SEE 3.7 + 3.3.14>

Add ACTIONS A, B, + G for Funct. 21 — M36

Add ACTIONS Note 1 for Function 21 — A28

Add ACTIONS NOTE 2 for Function 21 — A17

3.5.6 Accident Monitoring Instrumentation

A1

Applicability

Applies to accident monitoring instrumentation.

Objective

To ensure that sufficient information is available on selected plant parameters to monitor and assess such parameters following an accident.

Specifications

A1

~~3.5.6.1~~ LCO The accident monitoring instrumentation shown in Table 3.5.6-1 shall be operable per applicability indicated in the Table. The provisions of Technical Specification 3.0 do not apply. A29

3.5.6.2 RA G.1 In the event that the number of accident monitoring instrumentation channels falls below the limit given in Table 3.5.6-1 Column A; operation shall be limited as specified in Column B. A18

Bases

A2

The operability of the accident monitoring instrumentation for accident conditions as appropriate ensures that sufficient information is available on selected plant parameters to monitor and assess these variables following an accident.

RCS subcooled margin is directly indicated in the control room. Core subcooled margin is indicated on both ICC plasma displays, the OAC video, and a digital control board meter. Loop A subcooled margin is indicated on one ICC plasma display, the OAC video, and a digital control board meter. Loop B subcooled margin is indicated on the other ICC plasma display, the OAC video, and a digital control board meter. The OAC video and the digital control board meters are redundant displays of the same signal.

The operability requirements of the Reactor Coolant System subcooling margin monitors ensures that sufficient information is available to the operators to provide prompt recognition of saturated conditions in the primary coolant system and advanced warning of the approach to inadequate core cooling. Guidance for these requirements was provided by the NRC letter of July 2, 1980, and derived from the implementation of the TMI-2 lessons learned program.

Temperature indications from all 24 qualified core exit thermcouples can be displayed on the OAC. 12 qualified core exit thermcouples per train will input to each train of process electronics and can be displayed on the respective ICC plasma display.

Add ACTIONS NOTE 1 for PAM Functions 1, 3, 5, 6, 7, 9, 10, 16 + 17 A28

Add ACTIONS NOTE 2 for PAM Functions 1, 3, 5, 6, 7, 9, 10, 16 + 17 A17

A1 <Except as marked>

Table ~~3.5.6-1~~ 3.3.8-1

POST ACCIDENT MONITORING INSTRUMENTATION

M27

	Instrument	(A) Required Operable Channels	(B) Action	(C) Applicability
7 ~~1.~~	Containment Pressure Monitor (PT-230, -231)	2 of 2	1	Above hot shutdown
6 ~~2.~~	Containment Water Level Monitor Wide Range (LT-90, -91)	2 of 2	2	Above hot shutdown
9 ~~3.~~	Containment High-Range Radiation Monitor (RIA-57, -58)	2 of 2	2	Above hot shutdown
10 ~~4.~~	Containment Hydrogen Monitor (MT-80, -81)	2 of 2	2	Above hot shutdown
3 ~~5.~~	Wide Range Hot Leg Level (RC-LT0123, RC-LT0124)	2 of 2	3	Above hot shutdown
5 ~~6.~~	Reactor Vessel Head Level (RC-LT0125, RC-LT0126)	2 of 2	3	Above hot shutdown
16 ~~7.~~	Qualified Core Exit Thermocouple Trains	2 of 2 (a)	2	Above hot shutdown
17 ~~8.~~	Subcooling Monitors	2 (b)	4	When RCS temperature is > 300°F
1 ~~9.~~	Wide Range Nuclear Instrumentation	2 of ~~4~~	5	Above hot shutdown

LA5 LA5 LA11 LA5 A18 A30 Note d MODES 1, 2 + 3

Add PAM Functions 2, 4, 8, 11, 12, 13, 15, 18, 19, + 20 including associated LCO, Applicability, Actions, SRs, Notes and Table entries

M26

Page 4 of 11

Specification 3.3.8

A1 <Except as marked>

Table 3.5.6-1 (CONTINUED)
ACCIDENT MONITORING INSTRUMENTATION

ACTIONS

L22 RA A.1 for #7 Add RA B.1 for #7

Action 1: If one channel is inoperable, the channel shall be restored to operable status within ~~7~~ 30 (L22) days, or the unit shall be in hot shutdown within the next 12 hours.

RA C.1 for #7 RA H.1 for #7 RA H.2 for #7

If two channels are inoperable, at least one channel shall be restored to operable status within ~~48 hours~~ 7 days (L22), or the unit shall be in ~~hot shutdown~~ MODE 3 (A1) within the next 12 hours and MODE 4 within 18 hrs. (M27)

Add RA I.1 for #9 (L23)

L23 RA A.1 for #6, 9, 10, 16 Add RA B.1 for #6, 9, 10, 16

Action 2: If one channel is inoperable, the channel shall be restored to operable status within 30 days, or the unit shall be in hot shutdown within the next 12 hours.

RA C.1 for #6, 9, 16 RA D.1 for #10 RA H.1 for #6, 10, 16

If two channels are inoperable, at least one channel shall be restored to operable status within ~~48 hours~~ 7 days for #6, 9, 16 & 72 hrs for #10 (L23), or the unit shall be in ~~hot shutdown~~ MODE 3 within the next 12 hours and MODE 4 within 18 hrs. RA H.2 for #6, 10 + 16 (M27)

RA A.1 for #3, 5 (A12) RA B.1 for #3, 5

Action 3: If one channel is inoperable, the channel shall be restored to operable status within ~~7~~ 30 (L21) days, or a report shall be submitted to the Commission within the next 30 days outlining the cause of the inoperability and the plans and schedule for restoring the channel to operable status. <SEE 5.0>

L21 RA C.1 for #3, 5 Add RA I.1 for #3, 5

If two channels are inoperable, at least one channel shall be restored to operable status within 7 days, or the unit shall be in hot shutdown within the next 12 hours.

RA A.1 for #17 L24 Add RA B.1 for #17

Action 4: If one of the required channels is inoperable, at least one channel shall be restored to operable status within 30 days or the unit shall be in hot shutdown within the next 12 hours and below 300°F within the next 24 hours.

RA C.1 for #17 RA H.1 + H.2 for #17

If two of the required channels are inoperable, at least one channel shall be restored to operable status within ~~48 hours~~ 7 days (L24) or the unit shall be in ~~hot shutdown~~ MODE 3 within the next 12 hours and ~~below 300°F~~ MODE 4 within the next ~~24~~ 6 hours. (M27)

A1 <Except as marked>

Table 3.5.6-1 (CONTINUED)
ACCIDENT MONITORING INSTRUMENTATION

1 of the required channels — A13

Action 5: RA A1 for #1 — RA B1 for #1 — A12

If 3 channels are inoperable, at least one of the inoperable channels shall be restored to operable status within 30 days, ~~or a report shall be submitted to the NRC within the following 14 days. The report shall outline the preplanned alternate method of monitoring, the cause for the inoperability, and the plans and schedule for restoring the instrumentation channel to operable status.~~ <SEE 5.0>

2 of the required channels — A13

RA C1 for #1 — If ~~4 channels~~ are inoperable, at least one channel shall be restored to operable status within 7 days ~~or the unit shall be placed in~~ ~~hot shutdown~~ MODE 3 within an additional 12 hours — RA H.1 for #1

Add RA H.2 for #1 — and MODE 4 within 18 hours — M27

~~NOTES~~

T 3.3.8-1 Note d — Required Channels for ITS Function 16

(a) 6 of 12 qualified core exit thermocouples ~~must~~ be operable per train for a train to be considered operable — A30

(b) Operable subcooling margin monitors must consist of:

1) One direct indication for 1 of 2 RCS hot legs and one direct indication for the core; ~~or~~

2) One direct indication for each RCS hot leg.

LA11

Amendment No. 223 (Unit 1)
Amendment No. 223 (Unit 2)
Amendment No. 220 (Unit 3)

Add SR note A19

A1 <Except as marked>

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

<SEE 3.3.3 + 4>

Channel Description	Check	Test	Calibrate		Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA		
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	BS(1)	NA	(1)	(1)	Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	BS	45 Days STB	MO(1)(2)	(1) (2)	Using incore instrumentation. Axial offset upper and lower chambers after each startup if not done previous week.
~~5.~~ Wide Range	31 days ~~BS~~(1)	PS	NA	(1)	When in service.
6. Source Range	BS(1)	PS	NA	(1)	When in service.
7. Reactor Coolant Temperature	BS	45 Days STB	RF		
8. High Reactor Coolant Pressure	BS	45 Days STB	RF		
9. Low Reactor Coolant Pressure	BS	45 Days STB	RF		
10. Flux-Reactor Coolant Flow Comparator	BS	45 Days STB	RF		
11. Reactor Coolant Pressure Temperature Comparator	BS	45 Days STB	RF		

<SEE 3.3.1>

L26 SR 3.3.8.1

<SEE 3.3.9, 10> + 3.9.2

<SEE 3.3.1>

Add SR 3.3.8.3 for Function 1 M29

Specification 3.3.8

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(A1) <except as marked>

~~TABLE 4.1-1 (Continued)~~

~~Channel Description~~	SR 3.3.8.1 ~~Check~~	Test	SR 3.3.8.3 ~~Calibrate~~	~~Remarks~~
20. Reactor Building Spray System Logic	NA	MO	NA	SEE 3.3.5 + 7
21. Reactor Building Spray System Analog Channel - Reactor Building High Pressure	NA	MO	RF	
22. Pressurizer Temperature	ES	NA	RF	
23. Control Rod Absolute Position	ES(1)	NA	RF(2)	<SEE. 3.1> (1) Check with Relative Position Indicator. (2) Calibrate rod misalignment channel.
24. Control Rod Relative Position	ES(1)	NA	RF(2)	(1) Check with Absolute Position Indicator. (2) Calibrate rod misalignment channel.
25. Core Flood Tanks				
a. Pressure	ES	NA	RF	SEE 3.3.5
b. Level	ES	NA	RF	
11 ~~26.~~ Pressurizer Level SR 3.3.8.1	~~ES~~ 31 days (L26)	NA SR 3.3.8.3	~~RF~~ 18 months (A1)	
27. Letdown Storage Tank Level	DA	NA	RF	
28. Delete				
18 + 19 ~~29.~~ High and Low Pressure Injection Systems Flow Channels	~~NA~~ Add SR 3.3.8.1 (M26)	NA SR 3.3.8.3	~~RF~~ 18 months (A9)	

4.1-5

A 125/125/122
1/16/84

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Specification 3.3.8

(A1) <Except as marked>

~~Table 4.1-1 (CONTINUED)~~

~~Channel Description~~	SR 3.3.8.1 ~~Check~~ (L26)	Test	SR 3.3.8.3 ~~Calibrate~~	~~Remarks~~
14 ~~30~~. Borated Water Storage Tank Level Indicator	~~WE~~ 31 days	NA	~~RF~~ 18 months (A9)	
31. Boric Acid Mix Tank:				
a. Level	NA	NA	AN	(R1)
b. Temperature	MO	NA	AN	
32. Concentrated Boric Acid Storage Tank:				
a. Level	NA	NA	AN	
b. Temperature	MO	NA	AN	
33. Containment Temperature	NA	NA	RF (R1)	<SEE 3.2>
34. Incore Neutron Detectors	MO(1)	NA	NA	(1) Check functioning; including functioning of computer readout or recorder readout.
35. Emergency Plant Radiation Instruments	MO(1)	NA	RF	(1) Battery check. (R1)
36. Environmental Monitors	MO(1)	NA	RF	(1) Check functioning.
37. Reactor Manual Trip	NA	PS	NA <SEE 3.3.2>	
38. Reactor Building Emergency Sump Level	NA	NA	RF (R1)	
12 ~~39~~. Steam Generator Water Level	~~WE~~ 31 days (L26)	NA	~~RF~~ 18 months (A9)	
40. Turbine Overspeed Trip	NA	NA	RF (R1)	

12/11/87
Amendment No. 165 (Unit 1)
Amendment No. 165 (Unit 2)
Amendment No. 162 (Unit 3)

Specification 3.3.8

(A1) <Except as marked>

~~Table . . . .~~ (CONTINUED)

Channel Description		SR 3.3.8.1 ~~Check~~	Test	SR 3.3.8.3 ~~Calibrate~~	Remarks
21 ~~49~~	Emergency Feedwater Flow Indicators	~~MO~~ 31 days (A9)	NA	~~RF~~ 18 mo (A9)	
50.	PORV and Safety Valve Position Indicators	MO	NA	RF	
51.	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	RF	<SEE 3.3.1>
52.	RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
	a) Control Oil Pressure Switches	NA	45 Days STB	RF	
53.	Emergency Feedwater Initiation Circuits				
	a) Control Oil Pressure Switches	NA	MO	RF	<SEE 3.3.14>
9 ~~54.~~	Containment High Range Radiation Monitor (RIA-57, 58) (LA5)	~~NA~~ Add SR 3.3.8.1 (M33)	~~MO~~ (L27)	SR 3.3.8.3 ~~RF~~ 18 months (A9)	~~TMI Item II.F.1.3~~ (A24)

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Specification 3.3.8

Amendment No. 216
Amendment No. 216
Amendment No. 213

A1 <except as marked>

Table 4.1-1 (CONTINUED)

Channel Description	SR 3.3.8.1 ~~Check~~	Test	SR 3.3.8.2/3 ~~Calibrate~~	Remarks
7 ~~55.~~ Containment Pressure Monitor (PT-230, 231) LA5	~~MO~~ 31 days	NA	~~AN~~ 12 mo	TMI Item II.F.1.4 (A24)
6 ~~56.~~ Containment Water Level Monitor-Wide Range (LT-90, -91)	~~MO~~ 31 days	NA	~~RF~~ 18 mo	TMI Item II.F.1.5
10 ~~57.~~ Containment Hydrogen Monitor (MT-80, -81)	M33 ~~NA~~ 31 days	~~MO~~ L27	~~AN~~ 12 mo	TMI Item II.F.1.6
3 ~~58.~~ Wide Range Hot Leg Level	~~NA~~ 31 days	RF	~~RF~~ 18 mo.	A22
5 ~~59.~~ Reactor Vessel Head Level	~~NA~~ 31 days	RF	~~RF~~ 18 mo.	
16 ~~60.~~ Core Exit Thermocouples	~~MO~~ 31 days	NA	~~RF~~ 18 mo.	A9
17 ~~61.~~ Subcooling Monitors	~~MO~~ 31 days	RF	~~RF~~ 18 mo.	
62. Main Steam Header Pressure and MSLB detection (analog) channels	ES	RF	RF	<SEE 3.3.11>
63. Feedwater isolation circuitry (digital) channels and manual pushbutton	NA	RF	NA	<SEE 3.3.12 + 13>

A9

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage
STB - STAGGERED TEST BASIS

A9

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Specification 3.3.8

Specification 3.3.9

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operation Safety Instrumentation

A1

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

MODES 2, 3, 4 + 5 — M12 — L13

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7. — A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D. — A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light. (SEE 3.3.1)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved. (SEE 3.3.10)

Add RA A.1
Add RA B.1, B.2, B.3 + B.4
M13

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS

LA3

FUNCTIONAL UNIT	(A) TOTAL NO. OF CHANNELS	(B) CHANNELS TO TRIP	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions of Column C Cannot Be Met
1. Nuclear Instrumentation Wide Range Channels	4	NA	2	Bring to hot shutdown within 12 hours (b)
~~2.~~ Nuclear Instrumentation Source Range Channels	4	NA	2	Bring to hot shutdown within 12 hours (b) (c)
3. RPS Manual Pushbutton	1	1	1	Bring to hot shutdown within 12 hours
4. RPS Power Range Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
5. RPS Reactor Coolant Temperature Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
6. RPS Pressure-Temperature Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
7. RPS Flux Imbalance Flow Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
8. RPS Reactor Coolant Pressure				
a. High Reactor Coolant Pressure Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Low Reactor Coolant Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
9. RPS Power-Number of Pumps Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours (h)

LA3 LCO L28

<SEE 3.3.10>

<SEE 3.3.2>

<SEE 3.3.1>

Specification 3.3.9

Amendment No. 223 (Unit 1)
Amendment No. 223 (Unit 2)
Amendment No. 220 (Unit 3)

Pg 2 of 4

Specification 3.3.9

A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

5% L13

(b) Applic. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) RAC.1 When 2 of 4 wide range instrument channels are greater than 4×10^{-4} % rated power, hot shutdown is not required

Initiate action to restore required source range channel(s) within 1 hour M39

~~(d) (Deleted)~~

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours.

<SEE 3.3.5, 6, 7 + 15>

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state. <SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

Specification 3.3.9

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Table 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

Channel Description	Check	Test	Calibrate		Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA		
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1)	Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) (2)	Using incore instrumentation. Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	ES(1)	PS	NA	(1)	When in service.
6. Source Range	~~ES(1)~~ 12 hours	~~PS~~	~~NA~~	(1)	~~When in service.~~
7. Reactor Coolant Temperature	~~ES~~	~~45 Days STB~~	~~RF~~		
8. High Reactor Coolant Pressure	ES	45 Days STB	RF		
9. Low Reactor Coolant Pressure	ES	45 Days STB	RF		
10. Flux-Reactor Coolant Flow Comparator	ES	45 Days STB	RF		
11. Reactor Coolant Pressure Temperature Comparator	ES	45 Days STB	RF		

<SEE 3.3.3 + 3.3.4>

<See 3.3.1>

<SEE 3.3.8 + 10>

SR 3.3.9.1

A9

A1

Add SR 3.3.9.2

M14

L32

<SEE 3.3.1>

Amendment No. 223 (Unit 1)
Amendment No. 223 (Unit 2)
Amendment No. 220 (Unit 3)

M16

MODE 2, MODES 3, 4 + 5 with any CRD trip breaker in the closed position and the CRD System is capable of withdrawal

3.5 INSTRUMENTATION SYSTEMS

A1

3.5.1 Operation Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

L13

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met. With the exception of items 20, 21, and 22. For items 20, 21, and 22, the requirements are specified in Specification 3.5.7.

A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D.

A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light.

SEE 3.3.1

if not performed within the previous 7 days

L30

SR 3.3.10.3

3.5.1.5 During startup, when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved.

RA B.1
RA B.2

suspend positive reactivity additions + open CRD trip breakers within 1 hour

M15

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Amendment No. _____ (Unit 1)
Amendment No. _____ (Unit 2)
Amendment No. _____ (Unit 3)

A1 <except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS

LA3 M15 LCO L31

FUNCTIONAL UNIT	(A) ~~TOTAL NO. OF CHANNELS~~	(B) ~~CHANNELS TO TRIP~~	(C) ~~MINIMUM CHANNELS OPERABLE~~	(D) ~~Operator Action If Conditions of Column C Cannot Be Met~~
1. Nuclear Instrumentation Wide Range Channels	4	NA	2	~~Bring to hot shutdown within 12 hours (b)~~
2. Nuclear Instrumentation Source Range Channels	4	NA	2	Bring to hot shutdown within 12 hours (b) (c)
3. RPS Manual Pushbutton	1	1	1	Bring to hot shutdown within 12 hours
4. RPS Power Range Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
5. RPS Reactor Coolant Temperature Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
6. RPS Pressure-Temperature Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
7. RPS Flux Imbalance Flow Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
8. RPS Reactor Coolant Pressure a. High Reactor Coolant Pressure Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
b. Low Reactor Coolant Pressure Channels	4	2	3(a)	Bring to hot shutdown within 12 hours
9. RPS Power-Number of Pumps Instrument Channels	4	2	3(a)	Bring to hot shutdown within 12 hours (h)

Add RA B.1 + B.2

<SEE 3.3.9>

<SEE 3.3.2>

<SEE 3.3.1>

Specification 3.3.10

Amendment No. 223 (Unit 1)
Amendment No. 223 (Unit 2)
Amendment No. 220 (Unit 3)

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A1 <Except as marked>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

5% L13

(b) Applic When 2 of 4 power range instrument channels are greater than ~~10%~~ rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4 x 10^{-4} % rated power, hot shutdown is not required.

<SEE 3.3.9>

~~(d) (Deleted)~~

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours.

<SEE 3.3.5, 6, 7 + 15>

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state.

<SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

Oconee Units 1,2,and 3

4.1-3

Table 4.1-1
INSTRUMENT SURVEILLANCE REQUIREMENTS

<SEE 3.3.3 + 3.3.47>

Channel Description	Check	Test	Calibrate		Remarks
1. Protective Channel Coincidence Logic in the Reactor Trip Modules	NA	MO	NA		
2. Control Rod Drive Trip Breaker, SCR Control Relays E and F	NA	MO(1)	NA	(1)	This test shall independently confirm the operability of the shunt trip device and the undervoltage device.
3. Power Range Amplifier	ES(1)	NA	(1)	(1)	Heat balance check each shift. Heat balance calibration whenever indicated core thermal power exceeds neutron power by more than 2 percent.
4. Power Range	ES	45 Days STB	MO(1)(2)	(1) (2)	Using incore instrumentation. Axial offset upper and lower chambers after each startup if not done previous week.
5. Wide Range	SR 3.3.10.1 A9 ~~ES(1)~~ 12 hrs	L32 ~~PS~~	NA Add SR 3.3.10.2	M14 ~~(1)~~	~~When in service.~~ A1
6. Source Range	ES(1)	PS	NA	(1)	When in service.
7. Reactor Coolant Temperature	ES	45 Days STB	RF		
8. High Reactor Coolant Pressure	ES	45 Days STB	RF		
9. Low Reactor Coolant Pressure	ES	45 Days STB	RF		
10. Flux-Reactor Coolant Flow Comparator	ES	45 Days STB	RF		
11. Reactor Coolant Pressure Temperature Comparator	ES	45 Days STB	RF		

<See 3.3.1>

<SEE 3.3.9>

<SEE 3.3.1>

Amendment No. 223 (Unit 1)
Amendment No. 223 (Unit 2)
Amendment No. 220 (Unit 3)

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Specification 3.3.10

Specification 3.3.11

A1 <Except as marked>

except when all MFCVs and SFCVs are closed

L18

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

Applicability MODES 1 + 2 and MODE 3 — A16

Applic Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F.

<SEE 3.7>

Objective

To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.

Specifications

LCO 3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22. <SEE 3.3.12 + 13>

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

a. A Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition within 12 hours, and be less than or equal to an RCS temperature of 250 °F in an additional 18 hours.

<SEE 3.7>

Bases

A2

The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner. The completion times are considered adequate given the low probability of a MSLB accident.

Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main

TSC 95-03

A1 <Except as marked>

LA3

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

RA B.2.2

OR Close all MFCVs and SFCVs

L18

LCO

FUNCTIONAL UNIT	(A) ~~TOTAL NO. OF CHANNELS~~	(B) ~~CHANNELS TO TRIP~~	(C) MINIMUM CHANNELS OPERABLE		(D) Operator Action if Conditions of Column C Cannot Be Met
					ACTION B
20. Main Steam Header Pressure and MSLB detection (analog) channels per steam generator	3	2	3 (k)	RA. B.1 RA B.2.1	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.
21. Feedwater isolation circuitry (digital) channels	2	1	2 (l)		Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.
22. Feedwater isolation circuitry (digital) channels manual pushbutton	2	1	2 (l)		Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.

<SEE 3.3.12>

<SEE 3.3.13>

Add ACTIONS Note — A27

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Specification 3.3.11

Amendment No. ______ (Unit 1)
Amendment No. ______ (Unit 2)
Amendment No. ______ (Unit 3)

Specification 3.3.11

A1 <Entire Page>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

<SEE 3.3.4>

NOTES:

(j) 1. With one SCR Control Relay inoperable in logic channel C or D, restore the inoperable SCR Control Relay to OPERABLE status in 48 hours or remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within the next hour.

2. With two or more SCR Control Relays inoperable in logic channel C or D, remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within one hour.

ACTION A

(k) Requirement of 3 channels can be met with one of three channels placed in trip. The affected channel shall be placed in trip within 4 hours of discovery.

(l) 1 of 2 digital channels or manual pushbutton can be disabled for up to 72 hours and still meet the requirements of this column.

<SEE 3.3.12 + 13>

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A1 <Except as marked>

~~Table 4.1-1 (CONTINUED)~~

Channel Description	Check	Test	Calibrate	Remarks
55. Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	RF	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80,-81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	RF	RF	
59. Reactor Vessel Head Level	NA	RF	RF	
60. Core Exit Thermocouples	MO	NA	RF	
61. Subcooling Monitors	MO	RF	RF	
~~62.~~ Main Steam Header Pressure and MSLB detection (analog) channels	SR 3.3.11.1 ~~ES~~ 12 hrs A9	~~RF~~ A15	SR 3.3.11.2 ~~RF~~ 18 months A9	
63. Feedwater isolation circuitry (digital) channels and manual pushbutton	NA	RF	NA	

<SEE 3.3.8>

<SEE 3.3.12 + 13>

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage
STB - STAGGERED TEST BASIS

A9

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Specification 3.3.11

Specification 3.3.12

A1 <Except as marked>

except when all MFCVs and SFCVs are closed

L18

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

Applicability MODES 1 + 2 and MODE 3 — A16

Applic Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F.

<SEE 3.7>

Objective

To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.

Specifications

LCO 3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22. <SEE 3.3.11 + 13>

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

a. A Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition within 12 hours, and be less than or equal to an RCS temperature of 250 °F in an additional 18 hours.

<SEE 3.7>

Bases

A2

The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner. The completion times are considered adequate given the low probability of a MSLB accident.

Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main

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A1 <Except as marked>

LA3

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

<SEE 3.3.11>

ACTION B

FUNCTIONAL UNIT	(A) ~~TOTAL NO. OF CHANNELS~~	(B) ~~CHANNELS TO TRIP~~	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action if Conditions of Column C Cannot Be Met
20. Main Steam Header Pressure and MSLB detection (analog) channels per steam generator	3	2	3 (k)	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.
21. Feedwater isolation circuitry (digital) channels	2	1	2 (l)	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.
LCO 22. Feedwater isolation circuitry (digital) channels manual pushbutton	~~2~~	~~1~~	2 (l)	RA B.1 Bring to hot shutdown within 12 hours RA B.2.1 and bring to less than 700 psig steam header pressure within an additional 6 hours.

LA3

<SEE 3.3.13>

RA B.2.2 OR Close all MFCVs and SFCVs L18

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Specification 3.3.12

(A1) <Entire Page>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

<SEE 3.3.4>

NOTES:

(j) 1. With one SCR Control Relay inoperable in logic channel C or D, restore the inoperable SCR Control Relay to OPERABLE status in 48 hours or remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within the next hour.

2. With two or more SCR Control Relays inoperable in logic channel C or D, remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within one hour.

<SEE 3.3.11>

(k) Requirement of 3 channels can be met with one of three channels placed in trip. The affected channel shall be placed in trip within 4 hours of discovery.

ACTION A (l) 1 of 2 digital channels or manual pushbutton can be disabled for up to 72 hours and still meet the requirements of this column.

<SEE 3.3.13>

(A1) <Except as marked>

~~Table 4.1-1 (CONTINUED)~~

Channel Description	Check	Test	Calibrate	Remarks
55. Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56. Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	RF	TMI Item II.F.1.5
57. Containment Hydrogen Monitor (MT-80, -81)	NA	MO	AN	TMI Item II.F.1.6
58. Wide Range Hot Leg Level	NA	RF	RF	
59. Reactor Vessel Head Level	NA	RF	RF	<SEE 3.3.8>
60. Core Exit Thermocouples	MO	NA	RF	
61. Subcooling Monitors	MO	RF	RF	
62. Main Steam Header Pressure and MSLB detection (analog) channels	ES	RF	RF	<SEE 3.3.11>
63. Feedwater isolation circuitry (digital) channels and manual pushbutton <SEE 3.3.13>	NA	SR 3.3.12.1 ~~RF~~ 18 months (A4)	NA	

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage
STB - STAGGERED TEST BASIS

(A9)

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Specification 3.3.12

Amendment No. ______ (Unit 1)
Amendment No. ______ (Unit 2)
Amendment No. ______ (Unit 3)

Specification 3.3.13

A1 <Except as marked>

except when all MFCVs and SFCVs are closed

L18

3.5.7 Main Steam Line Break Detection and Feedwater Isolation

Applicability MODES 1 + 2 and MODE 3 A16

Applic Applies to main steam line break (MSLB) detection and feedwater isolation circuitry when main steam header pressure is greater than 700 psig and to the Main Feedwater main and startup control (Main Feedwater control) valves when Reactor Coolant System temperature is greater than 250 °F.

<SEE 3.7>

~~Objective~~

~~To ensure availability of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves to protect against containment overpressurization during a MSLB inside containment.~~

~~Specifications~~

LCO 3.5.7.1 MSLB detection and feedwater isolation circuitry shall be operable per Table 3.5.1-1, Items 20, 21, and 22 <SEE 3.3.11 + 12>

3.5.7.2 The Main Feedwater control valves shall be operable.

3.5.7.2.1 The provisions of 3.5.7.2 may be modified as follows:

a. A Main Feedwater control valve in one or more flow paths may be inoperable provided the affected valve(s) are closed within 8 hours from discovery and verified closed once per 7 days.

b. If the required actions and associated completion time of 3.5.7.2.1.a cannot be met, the reactor shall be placed in a hot shutdown condition within 12 hours, and be less than or equal to an RCS temperature of 250 °F in an additional 18 hours.

<SEE 3.7>

~~Bases~~ A2

~~The operability requirements of the MSLB detection and feedwater isolation circuitry and Main Feedwater control valves ensure that containment overpressure protection is available during a MSLB accident inside containment. The specified completion times provide adequate time to take appropriate action to restore the operability of the MSLB detection and feedwater isolation circuitry and the Main Feedwater control valves, or, if necessary, sufficient time to reduce power in a controlled manner. The completion times are considered adequate given the low probability of a MSLB accident.~~

~~Analyses of the main steam line break accident have determined that the containment design pressure of 59 psig could be exceeded with continued feedwater flow into the reactor building. To prevent exceeding the containment design pressure, the MSLB detection and feedwater isolation circuitry is designed to trip both Main Feedwater pumps, isolate all main~~

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A1 <Except as marked>

LA3

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

<SEE 3.3.11>

FUNCTIONAL UNIT	(A) ~~TOTAL NO. OF CHANNELS~~	(B) ~~CHANNELS TO TRIP~~	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action if Conditions of Column C Cannot Be Met
20. Main Steam Header Pressure and MSLB detection (analog) channels per steam generator	3	2	3 (k)	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.
21. Feedwater isolation circuitry (digital) channels	~~2~~	~~1~~	2 (l)	ACTION B RA B.1 Bring to hot shutdown within 12 hours RA B.2.1 and bring to less than 700 psig steam header pressure within an additional 6 hours. RA B.2.2 OR Close all MFCVs and SFCVs
22. Feedwater isolation circuitry (digital) channels manual pushbutton	2	1	2 (l)	Bring to hot shutdown within 12 hours and bring to less than 700 psig steam header pressure within an additional 6 hours.

LCO

LA3

<SEE 3.3.12>

L18

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Specification 3.3.13

(A1) <Entire Page>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

<SEE 3.3.4>

NOTES:

(j) 1. With one SCR Control Relay inoperable in logic channel C or D, restore the inoperable SCR Control Relay to OPERABLE status in 48 hours or remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within the next hour.

2. With two or more SCR Control Relays inoperable in logic channel C or D, remove power from the CRD mechanisms supplied by the inoperable channel's SCR Control Relay within one hour.

(k) Requirement of 3 channels can be met with one of three channels placed in trip. The affected channel shall be placed in trip within 4 hours of discovery.

<SEE 3.3.11>

ACTION A (l) 1 of 2 digital channels or manual pushbutton can be disabled for up to 72 hours and still meet the requirements of this column.

<SEE 3.3.12>

(A1) <Except as marked>

~~Table 4.1-1 (CONTINUED)~~

	Channel Description	Check	Test	Calibrate	Remarks
55.	Containment Pressure Monitor (PT-230, 231)	MO	NA	AN	TMI Item II.F.1.4
56.	Containment Water Level Monitor-Wide Range (LT-90, -91)	MO	NA	RF	TMI Item II.F.1.5
57.	Containment Hydrogen Monitor (MT-80,-81)	NA	MO	AN	TMI Item II.F.1.6
58.	Wide Range Hot Leg Level	NA	RF	RF	
59.	Reactor Vessel Head Level	NA	RF	RF	
60.	Core Exit Thermocouples	MO	NA	RF	
61.	Subcooling Monitors	MO	RF	RF	
62.	Main Steam Header Pressure and MSLB detection (analog) channels	ES	RF	RF	
63.	Feedwater isolation circuitry (digital) channels and manual pushbutton	NA	SR 3.3.13.1 ~~RF~~ 18 months	NA	

<SEE 3.3.8>

<SEE 3.3.11>

<SEE 3.3.12>

(A9)

ES - Each Shift
DA - Daily
WE - Weekly
MO - Monthly
QU - Quarterly
AN - Annually
PS - Prior to startup, if not performed previous week
NA - Not Applicable
RF - Refueling Outage
STB - STAGGERED TEST BASIS

(A9)

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Specification 3.3.13

Amendment No. ________ (Unit 1)
Amendment No. ________ (Unit 2)
Amendment No. ________ (Unit 3)

A1 <Except as marked>

3.4 SECONDARY SYSTEM DECAY HEAT REMOVAL

Applicability

Applies to the secondary system requirements for removal of reactor decay heat.

Objective

To specify minimum conditions necessary to assure the capability to remove decay heat from the reactor core.

Specification

MODES 1, 2, + 3
MODE 4 when steam generator is relied upon for heat removal
M25

Applic. 3.4.1 The reactor shall not be heated above 250°F unless the following conditions are met:

LCO
a. Three emergency feedwater pumps (one steam driven pump capable of being driven from an operable steam supply system and two motor driven pumps) and associated manual initiation circuitry shall be operable.

<SEE 3.7> <SEE 3.3.8>

b. Two 100% emergency feedwater flow paths shall be operable. Each flow path shall have at least one flow indicator operable.

MODES 1 + 2 — A10

LCO, LCO Note + Applic 3.4.2 In addition to the requirements of 3.4.1, prior to criticality, the automatic initiation circuitry associated with loss of main feedwater pumps as sensed by low hydraulic oil pressure shall be operable.

LA7

3.4.3 During operation greater than 250°F, the provisions of 3.4.1 and 3.4.2 may be modified to permit the following conditions:

<SEE 3.7>

a. One motor driven emergency feedwater pump may be inoperable for a period of up to seven days. If the inoperable pump is not restored to operable status within 7 days, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

<SEE 3.7 + 3.3.8>

b. One turbine driven emergency feedwater pump or one emergency feedwater flow path may be inoperable for a period of up to 72 hours. If the inoperable pump or flow path is not restored to operable status within 72 hours the unit will be at hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

c. Two motor driven emergency feedwater pumps may be inoperable for a period of up to 12 hours. If at least one pump is not restored to operable status within 12 hours, the unit shall be brought to hot shutdown within an additional 12 hours and below 250°F in another 12 hours.

Add ACTION A — L16

<SEE 3.7>

ADD ACTION B + 2nd part of COND. B — A10

<SEE 3.7>

d. With three emergency feedwater pumps and/or both emergency feedwater flow paths inoperable, immediately initiate corrective action to restore at least one emergency feedwater pump and associated emergency feedwater flowpath to operable status. The unit shall be at hot shutdown within 12 hours and below 250°F in another 12 hours if one emergency feedwater pump and associated flowpath are not restored to operable status.

LCO Note

e. If an emergency feedwater pump is inoperable ~~due only to~~ automatic initiation circuitry as specified by 3.4.2, the additional provisions of 3.4.3 a, b, c, and d which require cooldown of the RCS do not apply. A1D Not required to be OPERABLE in MODES 3 and 4

<SEE 3.7> 3.4.4 The 16 main steam safety relieve valves shall be operable.

3.4.5 A minimum of 72,000 gallons of water per operating unit shall be available in the upper surge tank, condensate storage tank, and hotwell. A minimum of 6 ft. (=30,000 gal) shall be available in the upper surge tank.

3.4.6 The controls of the emergency feedwater system shall be independent of the Integrated Control System.

3.4.7 The main steam atmospheric dump valve flow path on each steam generator shall be operable.

a. One main steam atmospheric dump valve flow path may be inoperable for a period of 7 days. If the inoperable flow path is not restored to operable status within 7 days, the unit shall be brought to hot shutdown within an additional 12 hours and below 350°F in another 24 hours.

b. Both main steam atmospheric dump valve flow paths may be inoperable for a period of 24 hours. If one main steam atmospheric dump valve flow path is not restored to an operable status within 24 hours, the unit shall be brought to hot shutdown within an additional 12 hours and below 350°F in another 24 hours.

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A1 <Entire page>

~~Table . . . (CONTINUED)~~

~~Channel Description~~		~~Check~~	~~Test~~	~~Calibrate~~	Remarks
49	Emergency Feedwater Flow Indicators	MO	NA	RF	<See 3.3.8>
50.	PORV and Safety Valve Position Indicators	MO	NA	RF	
51.	RPS Anticipatory Reactor Trip System Loss of Turbine Emergency Trip System Pressure Switches	NA	45 Days STB	RF	<SEE 3.3.1>
52.	RPS Anticipatory Reactor Trip System Loss of Main Feedwater				
	a) Control Oil Pressure Switches	NA	45 Days STB	RF	
53.	Emergency Feedwater Initiation Circuits				
	a) Control Oil Pressure Switches	NA	SR 3.3.14.1 ~~MO~~ 31 days	SR 3.3.14.2 ~~RF~~ 18 months	A9
54.	Containment High Range Radiation Monitor (RIA-57, 58)	NA	MO	RF	TMI Item II.F.1.3

<SEE 3.3.8>

Add SR 3.3.1.14.1 for manual initiation circuit — M20

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Specification 3.3.14

Amendment No. 216
Amendment No. 216
Amendment No. 213

3.5 INSTRUMENTATION SYSTEMS

A1

3.5.1 Operation Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objective

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

M40

Specifications

MODES 1, 2 + 3 except when all TSVs are closed

L34

LCO + Applic

3.5.1.1 The reactor shall not be in a startup mode or in a critical state unless the requirements of Table 3.5.1-1, Column C are met, with the exception of Items 20, 21, and 22. For Items 20, 21, and 22, the requirements are specified in Specification 3.5.7.

A1

3.5.1.2 In the event that the number of protective channels operable falls below the limit given under Table 3.5.1-1, Column C; operation shall be limited as specified in Column D.

A1

3.5.1.3 For on-line testing or in the event of a protective instrument or channel failure, a key-operated channel bypass switch associated with each reactor protective channel may be used to lock the channel trip relay in the untripped state. Status of the untripped state shall be indicated by a light. Only one channel bypass key shall be accessible for use in the control room. Only one channel shall be locked in this untripped state or contain a dummy bistable at any one time.

3.5.1.4 For on-line testing or maintenance during reactor power operation, a key-operated shutdown bypass switch associated with each reactor protective channel may be used in conjunction with a key-operated channel bypass switch as limited by 3.5.1.3. Status of the shutdown bypass switch shall be indicated by a light.

(SEE 3.3.1)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall not be greater than that readable on the source range instruments until the one decade overlap is achieved.

(SEE 3.3.10)

<SEE 3.3.5, 6 + 7>

TABLE 3.5.1-1
INSTRUMENTS OPERATING CONDITIONS (cont'd)

LA3

FUNCTIONAL UNIT	(A) ~~TOTAL NO. OF CHANNELS~~	(B) ~~CHANNELS TO TRIP~~	(C) MINIMUM CHANNELS OPERABLE	(D) Operator Action If Conditions Of Column C Cannot Be Met
15. ESF Reactor Building Spray System				
a. Analog Reactor Building High Pressure Instrument Channel	3	2	3	Bring to hot shutdown within 12 hours (e)
b. Digital Logic Manual Pushbutton	2	1	2	Bring to hot shutdown within 12 hours (e)
c. Digital Logic Channels (7 and 8)	2	1	2	Bring to hot shutdown within 24 hours (e)
16. Turbine Stop Valves Closure	~~2~~	~~1~~	2	~~Bring to hot shutdown within 24 hours (e)~~
17. Protective Channel Coincidence Logic in the Reactor Trip Modules	4 logic channels; A, B, C, and D	AB or AD or BC or CD	4	See Note (f)
18. CRD Breakers	1 AC Breaker and 2 DC Breakers per trip system		1 AC Breaker and 2 DC Breakers per trip system	See Note (i)
19. SCR Control Relays E and F	4 SCR Control Relays per trip system		4 SCR Control Relays per trip system	See Note (j)

LA3 L36 M22 LCO

<SEE 3.3.3 + 4>

Add ACTION A L19

Add SR 3.3.15.1 M23

3.5-5b

A 148,148,145
8/20/86

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Specification 3.3.15

Specification 3.3.15

A1 <Except as marked>

TABLE 3.5.1-1

INSTRUMENTS OPERATING CONDITIONS (cont'd)

NOTES:

<SEE 3.3.1>

(a) For channel testing, calibration, or maintenance, the minimum of three operable channels may be maintained by placing one channel in bypass and one channel in the tripped condition, leaving an effective one out of two trip logic.

(b) When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

(c) When 2 of 4 wide range instrument channels are greater than 4 x 10^{-4}% rated power, hot shutdown is not required.

<SEE 3.3.9 + 10>

~~(d) (Deleted)~~

(e) If minimum conditions are not met within 48 hours after hot shutdown, the unit shall be in cold shutdown within 24 hours. M22 L36

(f) 1. Place the inoperable Reactor Trip Module output in the tripped condition within one hour or

2. Remove the power supplied to the control rod trip devices associated with the inoperable Reactor Trip Module within one hour.

<SEE 3.3.3>

~~(g) (Deleted)~~

(h) The RCP monitors provide inputs to this logic. For operability to be met either all RCP monitor channels must be operable or 3 operable with the remaining channel in the tripped state. <SEE 3.3.1>

(i) 1. The power supplied to the control rod drive mechanisms through the failed CRD Trip Breaker shall be removed within one hour or

2. With one of the CRD Trip Breaker diverse features (undervoltage or shunt trip device) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip in the next hour.

<SEE 3.3.4>

L35

Once each refueling outage prior to ALTERATIONS or movement of irradiated fuel assemblies within containment

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

<SEE 3.9>

A36

LA14

SR 3.3.16.2

LCO + Applic

During CORE ALTERATIONS and during movement of irradiated fuel assemblies with the RB

3.8.10 The reactor building purge system, including the radiation monitor, RIA-45, which initiates purge isolation, shall be tested and verified to be operable ~~immediately prior to refueling operations~~.

3.8.11 Irradiated fuel shall not be moved from the reactor until the unit has been subcritical for at least 72 hours.

<SEE 3.9>

3.8.12 Two trains of spent fuel pool ventilation shall be operable with the following exceptions:

a. With one train of spent fuel pool ventilation inoperable, fuel movement within the storage pool or crane operation with loads over the storage pool may proceed provided the operable spent fuel pool ventilation train is in operation and discharging through the Reactor Building purge filters.

<SEE 3.7>

b. With no spent fuel pool ventilation filter operable, suspend all operations involving movement of fuel within the storage pool or crane operations with loads over the storage pool until at least one train of spent fuel pool ventilation is restored to operable status.

c. This specification does not apply during reracking operations with no fuel in the spent fuel pool.

3.8.13 a. Prior to spent fuel cask movement in the Unit 1 and 2 spent fuel pool, spent fuel stored in the first 36 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 55 days.

b. Prior to spent fuel cask movement in the Unit 3 spent fuel pool, spent fuel stored in the first 33 rows of the pool closest to the spent fuel cask handling area shall be decayed a minimum of 70 days.

<SEE 3.7>

c. Prior to dry storage transfer cask movement in the Unit 1 and 2 spent fuel pool, spent fuel stored in the first 64 rows of the pool closest to the cask handling area shall be decayed a minimum of 65 days.

d. Prior to dry storage transfer cask movement in the Unit 3 spent fuel pool, all spent fuel stored in that pool shall be decayed a minimum of 57 days.

3.8.14 No suspended loads of more than 3000 lbm shall be transported over spent fuel stored in either spent fuel pool.

Add ACTION A

M18

Add SR 3.3.16.1
SR 3.3.16.3

M19

Specification 3.3.16

Table 4.1-2
MINIMUM EQUIPMENT TEST FREQUENCY

	Item	Test	Frequency	
	1. Control Rod Movement (1)	Movement of Each Rod	Monthly	<SEE 3.1>
	2. Pressurizer Safety Valves	Setpoint	Each Refueling(4)	<See 3.4>
	3. Main Steam Safety Valves	Setpoint	Each Refueling(4)	<See 3.7>
SR 3.3.16.2	4. Refueling System Interlocks(5)	Functional	Prior to Refueling	L35
	5. Main Steam Stop Valves (1)	Movement of Each Stop Valve	Monthly	<SEE 3.7>
	6. Reactor Coolant System (2) Leakage	Evaluate	Daily	<SEE 3.4>
	7. Condenser Circulating Water(6) Flow Test	Functional	Each Refueling	<SEE 3.7>
	8. High Pressure Service Water Pumps and Power Supplies	Functional	Monthly	
	9. Spent Fuel Cooling System	Functional	Prior to Refueling	
	10. High Pressure and Low (3) Pressure Injection System	Vent Pump Casings	Monthly and Prior to Testing	<SEE 3.5>
	11. Emergency Feedwater Pump Automatic Start and Automatic Valve Actuation Feature	Functional	Each Refueling	<SEE 3.7>
	12. Main Steam Atmospheric Dump Valves	Stroke Test	Each Refueling	

(1) Applicable only when the reactor is critical. <SEE 3.1 + 3.7>

(2) Applicable only when the reactor coolant is above 200°F and at a steady-state temperature and pressure.

<SEE 3.4>

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(3) Operating pumps excluded. <SEE 3.5>

<SEE 3.4 + 3.7>

(4) Number of safety valves to be ~~tested each refueling~~ shall be in accordance with ASME Codes Section XI. Article IWV-3511, such that each valve is tested at least once every 5 years.

SR 3.3.16.2

(5) Applicable only to the interlocks associated with the Reactor Building Purge System.

(6) Verification of the Emergency Condenser Circulating Water (ECCW) System function to supply siphon suction to the Low Pressure Service Water System shall be performed to ensure operability of the LPSW System.

<SEE 3.7>

Specification 3.3.17

A1 <Except as noted>

INSTRUMENTATION

3.3 ~~3.7~~ ~~ELECTRICAL POWER SYSTEMS~~

3.3.17 ~~3.7.3~~ Emergency Power Switching Logic (EPSL) Automatic Transfer Functions

LCO 3.3.17 ~~TS 3.7.3~~ Two channels of the EPSL Automatic Transfer Function shall be OPERABLE.

Applic
APPLICABILITY: ~~Above COLD SHUTDOWN~~ MODES 1, 2, 3 and 4 A23

ACTIONS
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One channel inoperable.	A.1 Restore channel to OPERABLE status.	24 hours	ACT A
B. Required Action~~s~~ and associated Completion Time~~s~~ not met.	B.1 Be in ~~HOT SHUTDOWN~~ MODE 3 AND B.2 Be in ~~COLD SHUTDOWN~~ MODE 5	12 hours 84 hours	ACT B

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.3.1~~ 3.3.17.1 Perform ~~SR 3.7.1.14 (EPSL automatic transfer)~~ CHANNEL FUNCTIONAL TEST M28	~~As specified in applicable SR.~~ 18 months

Specification 3.3.17

<SEE 3.8>

<SEE 3.3.22>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.1.11	Verify each Keowee Hydro Unit can: 1) Emergency start from each control room; 2) Attain rated speed and voltage within 23 seconds of an emergency start initiate; 3) Be synchronized to the grid and loaded at the maximum practical rate to a value equivalent to one Unit's safeguard loads plus two Unit's HOT SHUTDOWN loads.	Annually
SR 3.7.1.12	NOTE Not required to be met when the overhead electrical disconnects for the Keowee Hydro Unit associated with the underground emergency power path are open. Verify the ability of the Keowee Unit ACBs to close automatically to the underground path.	Annually
SR 3.7.1.13	NOTE Only required to be met when a Lee gas turbine is energizing the standby buses. Verify that a Lee gas turbine can be started and connected to the isolated 100kV dedicated line and carry the equivalent of a single Unit's maximum safeguard loads within one hour.	Refueling LA12
~~SR 3.7.1.14~~ SR 3.3.17.1	Perform ~~an automatic transfer of the Main Feeder Buses to the Startup Transformer, Standby Buses, and retransfer to the Startup Transformers.~~ CHANNEL FUNCTIONAL TEST	~~Refueling~~ 18 months A9
SR 3.7.1.15	NOTE Only required to be met during periods of commercial power generation using the Keowee Hydro Units. Verify the ability of the Keowee Hydro units to supply emergency power from the initial condition of commercial power generation.	Refueling M28

(continued)

Specification 3.3.18

A1 <Except as marked>

3.3 ~~3.7~~ ~~ELECTRICAL POWER SYSTEMS~~ INSTRUMENTATION

3.3.18 ~~3.7.4~~ Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits

LCO 3.3.18 ~~TS 3.7.4~~ Three channels of each of the following EPSL voltage sensing circuits shall be OPERABLE:

a ~~1.~~ Startup ~~Source;~~ Transformer
b ~~2.~~ Standby Bus 1;
c ~~3.~~ Standby Bus 2;
d ~~4.~~ ~~Normal Source.~~ Auxiliary Transformer

A23

NOTE

LCO Note 1 If both N breakers are open, ~~Normal Source~~ Auxiliary Transformer voltage sensing is not required.

Add LCO Note 2

Applic APPLICABILITY: ~~Above COLD SHUTDOWN~~ MODES 1, 2, 3, and 4

ADD MODES 5 and 6, During movement of irradiated fuel assemblies M34

Actions ACTIONS

NOTE

ACT NOTE Separate Condition entry is allowed for each inoperable Voltage Sensing Circuit.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel of one or more circuits inoperable.	A.1 Restore channel to OPERABLE status.	24 hours ACT A
B. Required Action~~s~~ and associated Completion Time~~s~~ not met in MODES 1, 2, 3 + 4	B.1 Be in ~~HOT SHUTDOWN~~ MODE 3 AND B.2 Be in ~~COLD SHUTDOWN~~ MODE 5	12 hours ACT B 84 hours
C. Two or more channels of a required circuit inoperable when not in MODES 1, 2, 3 + 4. OR Required Action and associated Completion Time not met when not in MODES 1, 2, 3 + 4.	C.1 Declare the affected AC power source(s) inoperable	Immediately M34

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.4.1~~ 3.3.18.1 Perform a CHANNEL Functional TEST.	~~Refueling~~ 18 months A9

D. Required Action and associated Completion Time not met during movement of irradiated fuel assemblies	D.1 Suspend movement of irradiated fuel assemblies	Immediately M34

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Specification 3.3.19

A1 <Except as marked>

230 KV Switchyard DGVP

3.3 ~~3.7~~ INSTRUMENTATION
ELECTRICAL POWER SYSTEMS

3.3.19 ~~3.7.6~~ Emergency Power Switching Logic (EPSL) Degraded Grid Voltage Protection (DGVP)

230 KV Switchyard

LCO 3.3.19 ~~TS 3.7.6~~ The following ~~EPSL~~ Degraded Grid Voltage Protection functions shall be OPERABLE:

1. Three Switchyard Degraded Grid Voltage Sensing ~~Relays;~~ Channels and
2. Two channels of Switchyard Degraded Grid Voltage Protection Actuation Logic. Channels

Applic
APPLICABILITY: Above COLD SHUTDOWN MODES 1, 2, 3 and 4 — A23

Actions
ACTIONS

Place channel in trip. — L17

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One voltage sensing ~~relay~~ channel inoperable.	A.1 Restore voltage sensing relay to OPERABLE status.	72 hours	ACT A
B. One channel of actuation logic inoperable.	B.1 Restore channel to OPERABLE status.	72 hours	ACT B
C. Required Actions and associated Completion Times not met for Conditions A or B.	C.1 Be in ~~HOT SHUTDOWN~~ MODE 3 AND C.2 Be in ~~COLD SHUTDOWN~~ MODE 5	12 hours 84 hours	ACT C
D. Two or more voltage sensing ~~relays~~ channels inoperable. OR Two actuation logic channels inoperable.	D.1 Declare overhead emergency power path inoperable.	Immediately	ACT D

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.6.1~~ 3.3.19.2 Perform a CHANNEL CALIBRATION of the voltage sensing channel with setpoint Allowable Value as follows: Degraded voltage ≥ 219 kV and ≤ 222 kV with a time delay of 9 seconds ± 1 second.	~~Refueling~~ 18 months A9
SR ~~3.7.6.2~~ 3.3.19.1 Perform a CHANNEL FUNCTIONAL TEST.	~~Refueling~~ 18 months A9

Specification 3.3.20

A1 <Except as marked>

DGVP

3.3 ~~3.7~~ ELECTRICAL POWER SYSTEMS

3.3.20 ~~3.7.7~~ Emergency Power Switching Logic (EPSL) CT-5 Degraded Grid Voltage Protection (DGVP)

LCO 3.3.20 ~~TS 3.7.7~~ The following EPSL CT-5 Degraded Grid Voltage Protection functions shall be OPERABLE:
1. Three CT-5 Degraded Grid Voltage Sensing ~~Relays:~~ Channels and
2. Two channels of CT-5 Degraded Grid Voltage Protection Actuation Logic.

channels

Applic

MODES 1, 2, 3 and 4 — A23

APPLICABILITY: Above COLD SHUTDOWN when the Central switchyard is energizing the standby buses.

ACTIONS

ACTIONS

Place channel in trip. — L17

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One voltage sensing ~~relay~~ channel inoperable.	A.1 Restore voltage sensing relay to OPERABLE status.	72 hours	ACT A
B. One channel of actuation logic inoperable.	B.1 Restore channel to OPERABLE status.	72 hours	ACT B
C. Two actuation logic channels inoperable. OR Two or more voltage sensing channels ~~relays~~ inoperable. OR Required Actions and associated Completion Times cannot be met for Conditions A or B.	C.1 Open SL breakers.	1 hour	ACT C

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Specification 3.3.20

A1 <Except as marked>

DGVP

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR ~~3.7.7.1~~ 3.3.20.2 Perform a CHANNEL CALIBRATION of the voltage sensing channel with setpoint Allowable Value as follows: a. Degraded voltage ≥ 4143 V and ≤ 4185 V with a time delay of 9 seconds ± 1 second for the first level undervoltage inputs. b. Degraded voltage ≥ 3871 V and ≤ 3901 V for the second level undervoltage inputs.	~~Refueling~~ 18 months - A9
SR ~~3.7.7.2~~ 3.3.20.1 Perform a CHANNEL Functional TEST.	~~Refueling~~ 18 months - A9

3.3 ~~3.7~~ INSTRUMENTATION ~~ELECTRICAL POWER SYSTEMS~~

3.3.21 ~~3.7.5~~ Emergency Power Switching Logic (EPSL) Keowee Emergency Start Function

LCO 3.3.21 ~~TS 3.7.5~~ Two channels of the EPSL Keowee Emergency Start Function shall be OPERABLE.

Applic
APPLICABILITY: ~~Above COLD SHUTDOWN~~ MODES 1, 2, 3 and 4 A23

ACTIONS
ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours	Act A
B. Required Actions and associated Completion Times for Condition A not met.	B.1 Be in ~~HOT SHUTDOWN~~ MODE 3 AND B.2 Be in ~~COLD SHUTDOWN~~ MODE 5	12 hours 84 hours	Act B
C. Two channels inoperable.	C.1 Declare both Keowee Hydro Units inoperable ~~for the affected Oconee Unit(s)~~.	Immediately	Act C

SURVEILLANCE REQUIREMENTS

L33 LA12

	SURVEILLANCE	FREQUENCY
SR ~~3.7.5.1~~ 3.3.21.1	Perform ~~SR 3.7.1.11 (Keowee emergency start) and SR 3.7.1.14 (EPSL automatic transfer)~~. CHANNEL FUNCTIONAL TEST M28	~~As specified in applicable SR.~~ 18 months

Specification 3.3.21

<SEE 3.3.22>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.7.1.11 Verify each Keowee Hydro Unit can: 1) Emergency start from each control room; 2) Attain rated speed and voltage within 23 seconds of an emergency start initiate; 3) Be synchronized to the grid and loaded at the maximum practical rate to a value equivalent to one Unit's safeguard loads plus two Unit's HOT SHUTDOWN loads.	Annually
SR 3.7.1.12 ——————NOTE—————— Not required to be met when the overhead electrical disconnects for the Keowee Hydro Unit associated with the underground emergency power path are open. —————————————— Verify the ability of the Keowee Unit ACBs to close automatically to the underground path.	Annually
SR 3.7.1.13 ——————NOTE—————— Only required to be met when a Lee gas turbine is energizing the standby buses. —————————————— Verify that a Lee gas turbine can be started and connected to the isolated 100kV dedicated line and carry the equivalent of a single Unit's maximum safeguard loads within one hour.	LA12 Refueling
SR ~~3.7.1.14~~ 3.3.21.1 Perform ~~an automatic transfer of the Main Feeder Buses to the Startup Transformer, Standby Buses, and retransfer to the Startup Transformers~~ CHANNEL FUNCTIONAL TEST	~~Refueling~~ 18 months A9 M28
SR 3.7.1.15 ——————NOTE—————— Only required to be met during periods of commercial power generation using the Keowee Hydro Units. —————————————— Verify the ability of the Keowee Hydro units to supply emergency power from the initial condition of commercial power generation.	Refueling

(continued)

<SEE 3.8>

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Amendment _____ Unit 1
Amendment _____ Unit 2
Amendment _____ Unit 3

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Specification 3.3.22

Add LCO ACTIONS + SR for Specification 3.3.22 — M38

A1 <Except as marked>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR ~~3.7.1.11~~ 3.3.22.1 Verify each Keowee Hydro Unit can: 1) Emergency start from each control room; — LA13 2) Attain rated speed and voltage within 23 seconds of an emergency start initiate; 3) Be synchronized to the grid and loaded at the maximum practical rate to a value equivalent to one Unit's safeguard loads plus two Unit's HOT SHUTDOWN loads.	~~Annually~~ 12 months — A1 <SEE 3.8>
SR 3.7.1.12 ——NOTE—— Not required to be met when the overhead electrical disconnects for the Keowee Hydro Unit associated with the underground emergency power path are open. —— Verify the ability of the Keowee Unit ACBs to close automatically to the underground path.	Annually
SR 3.7.1.13 ——NOTE—— Only required to be met when a Lee gas turbine is energizing the standby buses. —— Verify that a Lee gas turbine can be started and connected to the isolated 100kV dedicated line and carry the equivalent of a single Unit's maximum safeguard loads within one hour.	Refueling
SR 3.7.1.14 Perform an automatic transfer of the Main Feeder Buses to the Startup Transformer. Standby Buses, and retransfer to the Startup Transformers.	Refueling — <SEE 3.3.17.1 + 3.3.21.1>
SR 3.7.1.15 ——NOTE—— Only required to be met during periods of commercial power generation using the Keowee Hydro Units. —— Verify the ability of the Keowee Hydro units to supply emergency power from the initial condition of commercial power generation.	Refueling — <SEE 3.8>

(continued)

ADMINISTRATIVE CHANGES

A1 Reformatting and renumbering are in accordance with NUREG-1430, Revision 1. As a result, the Technical Specifications should be more readily readable, and therefore understandable, by plant operators as well as other users. The reformatting, renumbering, and rewording process involves no technical changes to existing Technical Specifications.

Editorial rewording (either adding or deleting) is made consistent with NUREG-1430, Revision 1. During Improved Technical Specification (ITS) development certain wording preferences or English language conventions were adopted which resulted in no technical changes (either actual or interpretational) to the Technical Specifications. Additional information has also been added to more fully describe each subsection. This wording is consistent with NUREG-1430, Revision 1. Since the design is already approved by the NRC, adding more detail does not result in a technical change.

A2 The CTS Bases are completely replaced by revised bases that reflect the format and applicable content of proposed ITS Section 3.3. The revised Bases are shown in the proposed ITS Bases for Section 3.3.

A3 CTS 3.5.1.1 requires the reactor coolant system pressure parameter trip functions (≥ 1500 psig and ≥ 500 psig) to be OPERABLE when the reactor is in a startup mode or in a critical state. CTS 3.5.3 Notes (1) and (2) allow the reactor coolant system pressure parameter trip functions to be bypassed below 1750 and 900 psig respectively. ITS 3.3.5 Applicability for these two reactor coolant system pressure parameter trip functions is ≥ 1750 psig and ≥ 900 psig. The Notes of CTS 3.5.3 effectively change the applicability to the same as that proposed by ITS 3.3.5, therefore, the proposed change is considered administrative and consistent with the NUREG.

A4 CTS Table 2.3-1, Column "RPS Trip Setpoint" and "Shutdown Bypass" provide applicability requirements for RPS instrumentation equivalent to ITS Table 3.3.1-1 Notes a and b. CTS column "Shutdown Bypass" either states the function is bypassed or provides a different trip setting limit for RPS trips in Shutdown Bypass. The ITS 3.3.1-1 Table addresses the RPS trips by listing as separate functions, where appropriate, and with notes to the "Applicable Modes..." column of Table 3.3.1-1. These Notes are needed in the ITS due to the format differences between ITS and CTS to maintain allowances consistent with the use and application of the requirements of the corresponding portions of CTS Table 2.3-1. This change represents a change in presentation format only with no addition or deletion of requirements and is consistent with the NUREG.

A5 CTS Table 3.5.1-1 and Table 4.1-1 provide specific requirements for the power range instrument channels, as well as requirements for all the reactor trip functions which depend on these instruments for input with the exception of the nuclear overpower trip function. ITS 3.3.1,

including Table 3.3.1-1, deals individually with each RPS trip function which receives input from the power range instrument channels, including the nuclear overpower trip function. The power range nuclear instrument channels are not addressed separately by ITS. This is a change in the presentation of these requirements, with no actual change in requirements and is, therefore, administrative. The change is consistent with the NUREG.

A6 CTS 3.5.1.1 specifies that Table 3.5.1-1 Column C requirements be met for each functional unit. The required action when the minimum operable channel requirements of Column C are not met are prescribed separately for each functional unit. The ACTION Note of ITS 3.3.4, ITS 3.3.5, ITS 3.3.6, and ITS 3.3.7 allows separate condition entry for each parameter. This Note is needed in the ITS since the ACTIONS are not prescribed separately. This is a change in the presentation of requirements, with no actual change in requirements and is, therefore, administrative. The change is consistent with the NUREG.

A7 Not used.

A8 CTS Table 3.5.1-1, Column (D) and related Note (e) require the reactor to be in hot shutdown (equivalent to ITS MODE 3) in 12 hours and in Cold Shutdown (equivalent to ITS MODE 5) within 24 hours if the channel is not restored within 48 hours after hot shutdown. ITS 3.3.5 Required Action B.2.1 and B.2.2 is to reduce reactor coolant system pressure below the applicability, when one or more channels are inoperable or Required Action A.1 and associated Completion Time cannot be met. CTS 3.5.3 Notes (1) and (2) allow the reactor coolant system pressure parameter trip functions to be bypassed below 1750 and 900 psig respectively. Considering the allowance to bypass, the CTS action is essentially the same as the ITS ACTION. As such, the proposed change is considered administrative and consistent with the NUREG.

A9 CTS Table 4.1-1 Surveillance frequencies are replaced with those from the NUREG. The CTS and corresponding ITS Frequencies are as follows:

CTS	ITS
ES - Each shift	12 hours
DA - Daily	24 hours
WE - Weekly	7 days
MO - Monthly	31 days
QU - Quarterly	92 days
AN - Annually	12 months
PS - Prior to startup if not performed previous week	Not Used
NA - Not Applicable	Not Used
RF - Refueling Outage	18 months
Refueling	18 months
STB- STAGGERED TEST BASIS	STAGGERED TEST BASIS

Each of these changes is consistent with the current application of the CTS frequencies at ONS. These changes maintain requirements consistent

with both CTS and NUREG. These changes are administrative in nature because they represent a change in presentation format only with no change of actual requirements.

A10 CTS 3.4.2 requires the EFW System automatic initiation circuitry to be OPERABLE prior to criticality which is considered encompassed by ITS MODES 1 and 2. CTS 3.4.3.a through d capture required actions for EFW pumps inoperable due to inoperable EFW pump initiation circuitry. CTS 3.4.3.e excludes the requirement for RCS cooldown from those cases where the EFW pumps are inoperable only due to automatic initiation circuitry. These CTS provisions are captured by the Note to LCO 3.3.14. ITS 3.3.14, ACTION B is added to require the affected EFW pump to be declared inoperable immediately when its associated automatic or manual initiation circuitry is inoperable or when the Required Action and associated Completion Time of Condition A is not met. Allowed outage times for EFW System initiation circuitry are currently addressed indirectly as part of the EFW pump allowed outage times for CTS 3.4.3.a through d. ITS 3.3.14, ACTION B requires the affected EFW pumps to be declared inoperable immediately upon discovery that the associated automatic or manual initiation circuit is inoperable. As such, the ITS provides requirements equivalent to CTS 3.4.3.a through e.

A11 CTS 3.4.1 specifies that the reactor shall not be heated above 250°F unless each emergency feedwater flow path has an operable flow indicator. ITS 3.3.8 is applicable in MODES 1, 2, and 3. Since ITS MODE 3 is defined as ≥ 250°F, the applicabilities are equivalent. This minor difference (i.e., > versus ≥) is so close as to be imperceptible and is therefore considered administrative. The change is consistent with the NUREG.

A12 CTS Table 3.5.6-1 Action 3 and Action 5 require a Special Report be submitted when a PAM channel is inoperable for greater than 7 and 30 days respectively. ITS 3.3.8 Required Action B.1 requires action be initiated in accordance with ITS 5.6.6 immediately for the same condition. ITS 5.6.6 specifies requirements for this PAM Special Report. Changes to the Special Report requirements are addressed in the DOCs for Section 5.0. Therefore, the changes are due to presentation differences only and are therefore administrative. The change is consistent with the NUREG.

A13 CTS Table 3.5.6-1 Action 5 specifies required actions based on the number of channels available rather than the number required to be OPERABLE. ITS Table 3.3.8-1 list the number of channels required as two and the Conditions only apply to the number of required channels inoperable. There are 4 channels of Wide Range Nuclear Instrumentation available at ONS, however, only two channels are required to be OPERABLE for the PAM function. CTS Table 3.5.6-1 Action 5 requires action when 3 channels are inoperable (leaving one channel OPERABLE) and when 4 channels are inoperable (leaving no channels OPERABLE). Therefore, the requirements are the same although stated differently.

This change represents no actual change in requirements, only a change in presentation of requirements. This change is consistent with the NUREG.

A14 CTS 3.3.4 requires the BWST level instrumentation to be OPERABLE when the RCS, with fuel in the core, is in a condition with pressure equal to or greater than 350 psig or temperature equal to or greater than 250°F and subcritical. ITS 3.3.8 requires this instrumentation to be OPERABLE during MODES 1, 2, and 3. MODE 3 is defined as subcritical with the average coolant temperature > 250°F. CTS criteria specified as 250°F is considered more limiting than the 350 psig criteria, since the saturation temperature of water at 350 psig is > 435°F. As such, the proposed change is considered administrative. In addition, the CTS applicability statement "with fuel in the core" is deleted since the ITS definition of MODE is premised on "fuel in the vessel." This is a format change due to ITS conversion and is administrative in nature. The proposed change is consistent with the NUREG.

A15 The CTS Table 4.1-1 Item 62 specifies the testing requirements for the Main Steam Header Pressure and MSLB detection (analog) channels. These testing requirements are retained in ITS SR 3.3.11.2. As applied at ONS, CTS Table 4.1-1 requires a CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST of this function every refueling outage. Both the CTS and ITS definitions specify that the required calibration includes the CHANNEL FUNCTIONAL TEST. Therefore, the specific requirement to perform the CHANNEL FUNCTIONAL TEST is not retained in the ITS. This change represents no actual change in requirements, only a change in presentation of requirements and is consistent with the NUREG.

A16 CTS 3.5.7 Applicability requires the Main Steam Line Break (MSLB) detection and feedwater isolation circuitry to be OPERABLE when main steam header pressure is greater than 700 psig. ITS 3.3.11, 12, and 13 Applicability for this circuitry of MODES 1, 2, and MODE 3 when main steam header pressure is greater than 700 psig except when all MFCVs and SFCVs are closed. With the exception of "except when all MFCVs and SFCVs are closed," which is addressed in a separate less restrictive DOC, the CTS and ITS Applicabilities are equivalent. Therefore, the change is administrative and is consistent with the NUREG.

A17 CTS 3.5.6, 3.4.1, 3.4.3, and 3.3.4 provide requirements for Post Accident Monitoring (PAM) Functions. ITS 3.3.8 consolidates the CTS PAM Functions into one Specification. In CTS, each Function has separate Actions and is considered separately. However, since ITS addresses the PAM functions using common actions, Note 2 to the ACTIONS Table is used to indicate separate Condition entry is permitted for each function. This is a change in the presentation of requirements, with no actual change in requirements and is, therefore, administrative. The change is consistent with the NUREG.

A18 CTS 3.5.6.2 in conjunction with Table 3.5.6-1, Column B serves as a pointer to the appropriate action for each function (ITS PAM Functions

1, 3, 5, 6, 7, 9, 10, 16 and 17) when the number of instrument channels falls below the limit provided by Table 3.5.6-1, Column A. In ITS, the addition of ITS Table 3.3.8-1 Column "CONDITIONS REFERENCED FROM REQUIRED ACTION G.1" and Required Action G.1 provides comparable requirements. This is an administrative change only, and is necessary due to the different format used for ITS. This change is consistent with the NUREG.

A19 CTS Table 4.1-1 specifies instrumentation surveillance requirements for each instrument function separately. The ITS 3.3.8 SR Note is used to indicate that all 3.3.8 SRs apply to each PAM function in Table 3.3.8-1 except where indicated. The SR Note to SR 3.3.8.2 and SR Note 2 to SR 3.3.8.3 are provided to specify different calibration frequencies for Functions 7 and 10. The SR Notes are need only due to the change in presentation and format. This is an administrative change only, and is consistent with the NUREG.

A20 CTS 3.3.4 provides appropriate actions for an inoperable BWST level instrumentation channel separate from other PAM functions. ITS ACTION G is added for ITS Function 14 as a pointer to the appropriate action for this function. This is necessary because ITS 3.3.8 addresses the PAM functions using an instrumentation table and common actions, where appropriate. This is an administrative change only, and is necessary due to the different format used for ITS. This change is consistent with the NUREG.

A21 CTS Table 3.5.1-1 specifies the minimum channels required OPERABLE for CRD breakers and SCR (electronic trip assembly) control relays on a per trip system basis. ITS LCO 3.3.4 specifies the total number of channels required OPERABLE. Although presented differently, the two requirements are equivalent. As such, this change represents no actual change in requirements, only a change in presentation of requirements and is consistent with the NUREG.

A22 CTS Table 4.1-1 Items 58, 59, and 61 specifies the testing requirements for the Wide Range Hot Leg Level, Reactor Vessel Head Level, and Subcooling Monitor channels. These testing requirements are retained in ITS SR 3.3.8.3. As applied at ONS, CTS Table 4.1-1 requires a CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST of these functions every refueling outage. Both the CTS and ITS definitions specify that the required calibration includes the CHANNEL FUNCTIONAL TEST. Therefore, the specific requirement to perform the CHANNEL FUNCTIONAL TESTS is not retained in the ITS. This change represents no actual change in requirements, only a change in presentation of requirements and is consistent with the NUREG.

A23 The CTS 3.7.3 through 3.7.7 Applicability is above COLD SHUTDOWN. ITS 3.3.17 through 3.3.21 Applicability is MODES 1, 2, 3, and 4. CTS defines Cold Shutdown as RCS temperature $\leq 200°F$. Since ITS MODE 5 is defined as $\leq 200°F$, the Applicabilities are equivalent. As such, this

change represents no actual change in requirements, only a change in presentation of requirements and is consistent with the NUREG.

A24 The Remarks column of CTS Table 4.1-1 for Items 54, 55, 56, and 57 refer to TMI Items II.F.1.3, 4, 5, and 6 respectively. This reference to the specific origin of the CTS requirement for these post accident monitoring channels is for informational purposes and is deleted. The Bases for ITS 3.3.8, Post Accident Monitoring Instrumentation, provides appropriate reference to the origin of the requirements without calling out specific TMI Item numbers. This change represents no actual change in requirements, only a change in presentation of requirements and is consistent with the NUREG.

A25 Not used.

A26 Not used.

A27 CTS Table 3.5.1-1 and Note k requirements provide operability and action requirements based on instrument channels per steam generator. ITS 3.3.11 ACTIONS Note is added to allow separate condition entry for the main feedwater isolation function associated with a steam generator. This provision allows four hours to place a channel in trip for each function when Condition A is entered. The change is consistent with the CTS requirement and is only necessary due to the format of the ITS ACTIONS Table. In ITS, only four hours would be allowed from initial entry into the Table. Therefore, without the note, if another channel became inoperable sometime within the four hour period, only the time remaining from initial entry would be allowed. This change represents no actual change in requirements, only a change in presentation of requirements and is consistent with the NUREG.

A28 CTS does not preclude a change in mode while in an action statement. In general, ITS precludes MODE changes while relying on ACTIONS when the ACTIONS may eventually require a unit shutdown. ITS 3.3.8, ACTIONS Note 1 excludes the MODE change restrictions of LCO 3.0.4 for the PAMs. Therefore, the CTS and ITS allowances for mode changes for the PAM instruments are equivalent. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments. This change is consistent with the NUREG.

A29 CTS 3.5.6.1 excludes the provisions of Technical Specification 3.0 for the PAM instruments. ITS 3.3.8 does not provide a similar provision, however, none is needed since appropriate actions are provided for all conditions. Therefore, the change is administrative and consistent with the NUREG.

A30 CTS Table 3.5.6-1 specifies that 2 qualified core exit thermocouple trains must be OPERABLE. The accompanying Note (a) indicates that 5 of 12 qualified core exit thermocouples (CETs) must be operable per train for a train to be considered operable. ITS Table 3.3.8-1 specifies that the core exit temperature function consisting of 2 independent sets of 5 CETS shall be OPERABLE. Note d indicates that the subcooling margin monitor takes the average of the five highest CETs for each of the Inadequate Core Cooling Monitor (ICCM) trains. Although the presentation of these requirements differs, the requirements are equivalent. Therefore, the change is administrative and consistent with the NUREG.

A31 CTS Table 3.5.1-1 Note f.1 requires placing the Reactor Trip Module (RTM) output in the tripped condition within one hour of discovering it inoperable. ITS 3.3.3 Required Action A.1.1 requires tripping the associated CRD trip breaker within one hour. These two actions are considered equivalent since tripping the associated CRD trip breaker effectively places the RTM output in the tripped condition. Therefore, the change is administrative and consistent with the NUREG.

A32 CTS Table 3.5.1-1 Notes j.1 and j.2 require restoration of an inoperable SCR Control Relay (same as ETA relay) or removal of power from the CRD mechanisms supplied by the inoperable SCR Control Relay when one or more SCR control relays are inoperable. ITS 3.3.4 ACTION C requires tripping the corresponding AC CRD trip breaker(s)(Required Action C.2) when one or more ETA relays are inoperable. This action is equivalent since tripping the corresponding AC CRD trip breaker(s) removes power from the CRD mechanisms. Also, restoration is always an option in ITS so there is not need to provide a separate action for this. Therefore, the change is administrative and consistent with the NUREG.

A33 CTS Table 3.5.1-1, Item 12 describes the functional unit as ESF High Pressure Injection Systems and Reactor Building Isolation (Non-essential Systems. ITS LCO 3.3.6.a describes this function as High Pressure Injection, Reactor Building (RB) Non-Essential Isolation, Keowee Start, Load Shed and Standby Breaker Input and Keowee Standby Bus Feeder Breaker Input (ES Channels 1 and 2). The ITS LCO statement was expanded to provide a complete list of equipment actuated by the ES channels. The LCO statement does not change the technical specification requirements. Therefore, the change is administrative.

A34 CTS Table 3.5.1-1, Item 13 describes the functional unit as ESF Low Pressure Injection System. ITS LCO 3.3.6.b describes this function as the Low Pressure Injection System and Reactor Building Essential Isolation (ES Channels 3 and 4). The ITS LCO statement was expanded to provide a complete list of equipment actuated by the ES channels. The LCO statement does not change the technical specification requirements. Therefore, the change is administrative.

A35 CTS Table 3.5.1-1, Item 14 describes the functional unit as ESF Reactor Building Isolation (Essential Systems) and Reactor Building Cooling System. ITS LCO 3.3.6.c describes this function as Reactor Building (RB) Cooling, Reactor Building Essential Isolation and Penetration Room Ventilation (ES Channels 5 and 6). The ITS LCO statement was expanded to provide a complete list of equipment actuated by the ES channels. The LCO statement does not change the technical specification requirements. Therefore, the change is administrative.

A36 CTS 3.8.1.10 requires the reactor building purge system, including the radiation monitors, to be operable immediately prior to refueling operation. ITS 3.3.16 requires the Reactor Building Isolation-High Radiation function to be OPERABLE during CORE ALTERATION and during movement of irradiated fuel assemblies within the reactor building. CTS defines "refueling operation" as an operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed. Also, movement of irradiated fuel assemblies within the reactor building can only be performed subsequent to the start of refueling. As such, the change is administrative and consistent with the NUREG.

TECHNICAL CHANGE-MORE RESTRICTIVE

M1 CTS Table 3.5.1-1, Column (D) requires the operator to place the plant in hot shutdown (ITS equivalent of MODE 3) within 12 hours when the minimum channels OPERABLE requirement of Column (C) is not met. ITS provides an equivalent requirement (3.3.1 Required Action C.1, 3.3.2 Required Action B.1) and adds a requirement to open all CRD trip breakers within 12 hours (3.3.1 Required Action C.2, 3.3.2 Required Action B.2). The added requirement is appropriate since it places the unit in a condition where the LCO does not apply. The proposed change represents an additional restriction on unit operation and is consistent with the NUREG.

M2 CTS 3.5.1.1 requires the RPS functions of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. ITS 3.3.1 LCO Applicability for each RPS function is according to Table 3.3.1-1. While the ITS Applicability is equivalent for most functions, it is also more restrictive for some functions. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq 0.99\ k_{eff}$. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The ITS Applicability for the following RPS functions is considered more restrictive:

CTS does not explicitly require the Nuclear Overpower - Low Setpoint and the Shutdown Bypass RCS High Pressure to be OPERABLE when the reactor is shutdown with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. ITS 3.3.1 requires these functions to be OPERABLE in MODES 3, 4, and 5 during shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal (ITS Table 3.3.1-1, Note b). CTS 3.1.9, Low Power Physics Testing Restrictions, does require these functions to be OPERABLE during low power physics testing. However, the ITS Applicability of during shutdown bypass operations with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal is more restrictive since the CTS Applicability is limited to Physics Testing. This is considered appropriate since it ensures the capability to trip the withdrawn control rods exists at all times that rod motion is possible.

CTS requires the Reactor Building High Pressure function to be OPERABLE when the reactor is in a startup mode or in a critical state (equivalent to ITS MODES 1 and 2 as discussed above). ITS 3.3.1 requires this function to be OPERABLE during MODES 1 and 2 and MODE 3 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal (ITS Table 3.3.1-1 and Note c). The additional requirement for OPERABILITY in MODE 3 is considered appropriate to ensure that the instrumentation required to initiate the insertion of any withdrawn CONTROL RODS is OPERABLE whenever CONTROL

RODS are withdrawn or capable of withdrawal. The automatic insertion of any withdrawn CONTROL ROD is consistent with evaluations of accidents initiated from MODE 3.

CTS requires the Nuclear Overpower - High Setpoint and RCS High Pressure function to be OPERABLE when the reactor is in a startup mode or in a critical state (equivalent to ITS MODES 1 and 2 as discussed above). ITS 3.3.1 requires these functions to be OPERABLE during MODES 1 and 2 and MODE 3 when not in shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal (ITS Table 3.3.1-1, Note d). The additional requirement for OPERABILITY in MODE 3 is considered appropriate to ensure that the instrumentation required to initiate the insertion of any withdrawn CONTROL RODS is OPERABLE whenever CONTROL RODS are withdrawn or capable of withdrawal. The automatic insertion of any withdrawn CONTROL ROD is consistent with evaluations of accidents initiated from MODE 3. The proposed more restrictive changes to the Applicability for RPS functions are appropriate since they provide additional assurance that a reactor trip will be actuated if needed. These changes are consistent with the NUREG as modified by JFD 24.

M3 CTS 3.5.1.1 requires the RPS manual trip function, Reactor Trip Module, and CRD trip devices of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. The ITS applicability for the manual reactor trip function (ITS 3.3.2), the RTMs (ITS 3.3.3) and the CRD trip devices (ITS 3.3.4) is MODES 1 and 2 and MODES 3, 4, and 5 during shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq$ 0.99 k_{eff}. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. Since the ITS is applicable under more conditions than the current applicability, the ITS Applicability is more restrictive. The proposed applicability provides additional assurance that a reactor trip will be actuated if needed to prevent accident conditions from exceeding those calculated in the accident analyses. This change is consistent with the NUREG.

CTS Table 3.5.1-1 does not have any required actions for the manual reactor trip function, the RTMs and the CRD trip devices when in MODE 3 or lower since Column D only requires the unit be place in hot shutdown (equivalent to ITS MODE 3). Consistent with the proposed ITS applicability, required actions are provided for each ITS Specification that require the unit be placed in a condition outside the applicability of the Specification when the LCO Required Actions cannot be met. ITS 3.3.2 Required Action C.1 is added for the manual reactor trip function to require the operator to open all the CRD trip breakers in 6 hours. Appropriate Required Actions are provided for the RTM (ITS 3.3.3 Required Action C.1 and C.2) and CRD trip devices (ITS 3.3.4

Required Action E.1 and E.2) when they are inoperable in MODE 3 or lower. This additional restriction on operation is appropriate since it ensures rod motion is not possible when the required trip function or devices are inoperable. This change is consistent with the NUREG.

M4 CTS Table 3.5.1-1 Note f requires an inoperable RTM be placed in trip in 1 hour or remove power to the associated CRD trip breaker in 1 hour. ITS 3.3.3 Required Action A.2 is added to these actions to require that the inoperable RTM be removed from the cabinet. The addition of this action is appropriate since it ensures that the trip signal is registered in the other channels by causing the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. This additional restriction on operation is consistent with the NUREG.

M5 CTS Table 3.5.1-1 provides no specific requirements to the address the condition where the action specified for an inoperable RTM or CRD trip breaker is not completed within the required time period. ITS 3.3.3 ACTION B and ITS 3.3.4 ACTION D are added to require the Unit be in MODE 3 in 12 hours with all CRD trip breakers open or to require the operator to remove power from all CRD trip breakers when the Required Action and associated Completion time is not met in MODE 1, 2, or 3. For ITS 3.3.3 ACTION B this action also applies to the condition where two or more RTMs are inoperable in MODE 1, 2, or 3. CTS requires entry into CTS 3.0, which requires the reactor be place in Hot Shutdown (equivalent to ITS Mode 3) in 12 hours. The ITS ACTION is more appropriate since it places the unit in a condition in which the LCO no longer applies (i.e., also requires opening all CRD trip breakers or removing power from all CRD trip breakers). This change is consistent with the NUREG as modified by JFD 17.

M6 CTS 3.5.1.1 requires the ESPS functions of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. The ITS 3.3.5 Applicability for ESPS Parameters 3 and 4 is MODES 1, 2, 3, and 4. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is ≥ 0.99 k_{eff}. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. Since the ITS is applicable under more conditions than the current applicability, the ITS Applicability is more restrictive. The additional requirement for OPERABILITY of these functions in MODES 3 and 4 is considered appropriate since the potential for a high energy line break exists. This change is consistent with the NUREG.

M7 CTS Table 3.5.1-1 Note (e), in conjunction with CTS Table 3.5.1-1 Column D, provides a total time of 84 hours, from failure to meet the MINIMUM CHANNELS OPERABLE requirement of Column C for the ESPS parameters, for the unit to enter cold shutdown (equivalent it ITS

MODE 5). ITS 3.3.5 Required Action B.2.3 and ITS 3.3.6 Required Action B.2 require entry into MODE 5 within 36 hours of failure to meet the LCO. These more restrictive requirements minimize the time during which the safety function is degraded while providing sufficient time to accomplish an orderly shutdown. Additionally, this Completion Time is consistent with NUREG.

M8 CTS 3.5.1.1 requires the ESPS functions of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. The ITS 3.3.6 and 3.3.7 Applicability is MODES 1 and 2 and MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq 0.99\ k_{eff}$. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The ITS Applicability of MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE is more restrictive. These additional requirements are considered appropriate since the potential for a high energy line break exists. This change is consistent with the NUREG.

M9 CTS Table 2.3-1 does not include Allowable Values for the Main Turbine Trip (Hydraulic Fluid Pressure) function and the Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) function. ITS Table 3.3.1-1 provides an Allowable Value for each of these functions (ITS Functions 9 and 10, respectively). Their addition is appropriate since these functions provide an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip or loss of main feedwater. The turbine trip lowers the probability of an RCS PORV actuation for turbine trip cases. The loss of main feedwater trip provides a reactor trip at high power levels to minimize challenges to the PORV. Because these values are not specified in CTS, this change represents additional restrictions on unit operation. This change is consistent with the NUREG.

M10 CTS Table 3.5.1-1 Note (i)2 requires that with one of the CRD trip breaker diverse feature inoperable that it be restored within 48 hours or placed in trip within the next hour. ITS 3.3.4 Required Action A.1 is more restrictive since it requires the breaker to be tripped in 48 hours (eliminates the extra one hour). This reduction in total Completion Time from 49 hours to 48 hours is adopted to provide requirements consistent with the NUREG. The 48 hour Completion Time is adequate to perform the required actions.

M11 CTS Table 3.5.1-1 Note (j) allows up to 48 hours to restore a single inoperable SCR relay prior to requiring the power be removed from the CRD mechanisms supplied by the inoperable channel's SCR relay within the next hour. In the event more than one SCR relay in a channel is inoperable, all SCR trip devices in the channel are to be tripped within one hour. ITS 3.3.4 Required Action C.2 requires the

corresponding AC CRD trip breaker be tripped within one hour when one or more ETA relays are inoperable. This ACTION contains no provision for a 48 hour delay prior to requiring additional action to be taken with only one ETA relay inoperable. Because action to compensate for a single inoperable ETA Relay is required sooner by ITS than by CTS, this change is more restrictive. The reduced allowed outage time is considered reasonable in that it provides sufficient time to perform the Required Action. The proposed change is consistent with the NUREG.

M12 CTS Table 3.5.1-1, Column C requires the source range channels to be OPERABLE when the reactor is in a startup mode or in a critical state except as modified by Note b to the table which modifies the applicability to exclude operation at greater than 10% power. ITS 3.3.9 LCO Applicability is MODES 2, 3, 4, and 5. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq 0.99\ k_{eff}$. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The expanded applicability is appropriate since it ensures the operator is provided with a means of monitoring neutron flux and provides an early indication of reactivity changes. The modification of the applicability exclusion for operation above 10% to MODE 2 (above 5% power) is discussed in L13. The proposed change is consistent with the NUREG.

M13 CTS Table 3.5.1-1 (Column D action for RPS Functional Unit 2, source range instrument channels) requires the unit be placed in hot shutdown (equivalent to ITS MODE 3) within 12 hours when one or more required instrument channels are inoperable when the reactor is in a startup mode or in a critical state (equivalent to ITS MODES 1 and 2 as described in M12 above). CTS does not provide actions below hot shutdown. Because of the expanded Applicability described in DOC M12 above, ITS 3.3.9 ACTIONS A and B are added to provide appropriate actions in MODE 3 or below. With one channel inoperable when the THERMAL POWER level is $\leq 4 \times 10^{-4}$ RTP on the wide range neutron flux channels, ITS 3.3.9 Required Action A.1 requires the channel be restored prior to increasing thermal power. With two channels inoperable when the THERMAL POWER level is $\leq 4 \times 10^{-4}$ RTP on the wide range neutron flux channels, ITS 3.3.9 Required Action B.1 and B.2 requires immediate suspension of operations involving positive reactivity changes and the immediate initiation of action to insert all control rods. Required Actions B.3 and B.4 require the control rod drive trip breakers be opened within 1 hour and that SDM be verified $\geq 1\%\ \Delta k/k$ within 1 hour. These more restrictive ACTIONS provide requirements which ensure that the unit is placed in an acceptable condition to compensate for the inoperability of the required source range instrument channel. These additional ACTIONS are appropriate and consistent with NUREG requirements.

M14 CTS Table 4.1-1 Items 5 and 6 do not require a calibration of the source range and wide range instruments. ITS SR 3.3.9.2 and ITS SR 3.3.10.2 are added to require a CHANNEL CALIBRATION for these instruments on an 18 month Frequency. This test ensures the channel responds to measured parameters within the necessary range and accuracy and leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. The more restrictive change is consistent with NUREG.

M15 CTS 3.5.1-1 Column D requires the unit be place in hot shutdown within 12 hours when two required wide range instrument channels are inoperable. CTS 3.5.1.5 requires action be performed in the event one decade of overlap between the source range and wide range instruments is not achieved and specifies that the flux level not be greater than that readable on the source range instruments until the one decade overlap is achieved. This is replaced by ITS 3.3.10 ACTION B which requires operations involving positive reactivity changes to be suspended (Required Action B.1)and CRD trip breakers to be opened (Required Action B.2). These actions are appropriate since they place the unit in a condition outside the Applicability for the wide range instrumentation. The wide range instrumentation is designed to detect power changes during initial criticality and power escalation when the power range and source range instrumentation cannot provide reliable indications. Having both instruments inoperable could prevent the operator from detecting and controlling neutron flux transients that could result in a reactor trip during power escalation. The ITS ACTION presents more restrictive requirements in that unlimited continued operation in the source range will no longer be allowed. This change is consistent with NUREG.

M16 CTS 3.5.1.1 requires the wide range instruments of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. ITS 3.3.10 Applicability is MODE 2 and MODES 3, 4, 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq 0.99\ k_{eff}$. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The expanded applicability is appropriate since it provides assurance that wide range instrumentation will be available to detect power changes during initial criticality and power escalation when the power range and source range instrumentation cannot provide reliable indications. This change is consistent with the NUREG as modified by JFD 19. The deletion of the requirement for these instruments in MODE 1 is discussed in DOC L13.

M17 CTS Table 4.1-1, item 13 requires a channel check of the high reactor building pressure function daily. The interval for this check is shortened to 12 hours in ITS SR 3.3.1.1 consistent with the NUREG and

other RPS instruments. The proposed change is more restrictive since it is an additional restriction on operation and is made for consistency with the NUREG.

M18 CTS 3.8.10 requires the radiation monitor associated with the purge system to be tested and verified OPERABLE immediately prior to refueling operations. No explicit requirement is provided in the CTS that addresses required action if monitor is later discovered inoperable. Therefore, CTS LCO 3.0 would require a unit shutdown to MODE 5. Since the unit is already shutdown, no action would be required. ITS 3.3.16 ACTION A, which provides appropriate actions for this situation, is added and provides two options. Required Action A.1 requires the reactor building purge valves to be closed immediately or Required Action A.2 requires the movement of irradiated fuel assemblies to be suspended. Closure of the purge valves accomplishes the function of the high radiation channel. Suspending movement of irradiated fuel assemblies places the unit in a configuration in which the purge isolation on high radiation is not required. The proposed change is more restrictive since it is an additional restriction on plant operation and is consistent with the NUREG.

M19 CTS 3.8.10 does not provide a specific requirement to check or calibrate the reactor building purge valve isolation - high radiation channel. ITS SR 3.3.16.1 requires a CHANNEL CHECK of this channel every 12 hours and ITS SR 3.3.16.3 requires a CHANNEL CALIBRATION of this channel every 18 months. The addition of the CHANNEL CHECK is an appropriate restriction to ensure that a gross failure of instrumentation has not occurred. The CHANNEL CALIBRATION provides a complete check of the instrument loop and sensor and is an appropriate restriction to verify the channel responds to a measured parameter within the necessary range and accuracy. The changes are consistent with the NUREG.

M20 CTS Table 4.1-1 does not explicitly require a channel test of the EFW System manual initiation circuit. ITS SR 3.3.14.1 is added to provide an explicit requirement for a CHANNEL FUNCTIONAL TEST. The addition of the CHANNEL FUNCTIONAL TEST is appropriate to ensure that the manual initiation circuit can perform its intended function. This is an additional restriction on operation consistent with the NUREG.

M21 CTS Table 4.1-1, Item 21, Column "Check" currently does not require a check of the Reactor Building Pressure-High High parameter. ITS SR 3.3.5.1 is added to require a CHANNEL CHECK of the Reactor Building Pressure-High High parameter every 12 hours consistent with the NUREG. A CHANNEL CHECK provides reasonable assurance that a gross failure of instrumentation will be identified promptly. The more restrictive change is an acceptable restriction on operation and is consistent with the NUREG.

M22 CTS Table 3.5.1-1, Column D requires the unit to be in hot shutdown (MODE 3) within 24 hours when one or more TSV Closure Instrumentation

channels is inoperable and Note (e) to the Table requires the unit be placed in Cold Shutdown (MODE 5) within the following 72 hours if the minimum conditions are not met. ITS 3.3.15 ACTION A is added to require the TSVs to be declared inoperable within 1 hour (also, see DOC L19). ITS 3.7.2, Turbine Stop Valves, then dictates the required action for inoperable TSVs. With one or more TSVs inoperable in MODE 1, Required Action A.1 requires the TSVs be restored to OPERABLE status within 12 hours or Required Action B.1 requires the unit be in MODE 2 in 6 hours. Therefore, this portion of ITS is more restrictive since the unit must be in MODE 2 within 15 hours of an inoperable TSV Closure instrumentation channel where CTS required the unit be in hot shutdown (equivalent to ITS MODE 3) within 24 hours. ITS 3.7.2 Action C allows 8 additional hours to close an inoperable TSV when in MODE 2 or 3, which is slightly more restrictive since a total of 23 hours is allowed to close the TSV from initial discovery of it being inoperable in MODE 1. In addition, if it were not closed, then an additional 12 hours (on top of the eight hours) is allowed to place the unit in MODE 3 and 18 hours to place the unit in MODE 4. This results in allowing a total of 35 hours to be in MODE 3 and 41 hours to be in MODE 4 from initial discovery of it being inoperable in MODE 1. This compares to the CTS shutdown time of 24 hours to be in hot shutdown (MODE 3) and 96 hours to be in cold shutdown (MODE 5). The less restrictive aspects of this change are discussed in DOC L36. The proposed more restrictive ITS Shutdown Times requirements are considered reasonable since they are consistent with ITS 3.7.2 which is consistent with the NUREG.

These more restrictive changes provide reasonable assurance that the TSVs can perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits.

M23 CTS does not include any test requirements for the TSV Closure Instrumentation channels. ITS SRs 3.3.15.1 is added to require a CHANNEL FUNCTIONAL TEST be performed every 31 days to ensure that the channel can perform its intended function. This test is an appropriate restriction on unit operation. The more restrictive requirement is consistent with comparable NUREG (3.3.11-1, Function 4) requirements.

M24 CTS 3.3.4.a(2) requires the unit be in Hot Shutdown (equivalent to ITS MODE 3) within 12 hours and in a condition with RCS pressure below 350 psig and RCS temperature below 250°F (equivalent to ITS MODE 4) within an additional 48 hours when one required channel of BWST level instrumentation is inoperable for more than 24 hours. ITS 3.3.8 Required Actions H.1 and H.2 require the unit to be placed in ITS MODE 3 (i.e., subcritical) in 12 hours, and in ITS MODE 4 in 18 hours. The shorter Completion Time is reasonable to allow this MODE to be reached in an orderly manner and without challenging unit systems. The proposed change is consistent with the NUREG.

M25 CTS 3.4.1 requires the EFW pump initiation circuitry to be OPERABLE when Reactor Coolant System (RCS) temperature is > 250°F. ITS 3.3.14

APPLICABILITY for EFW pump initiation circuitry is MODES 1, 2, & 3 and MODE 4 when a steam generator is relied upon for heat removal. The ONS design precludes exceeding 246°F except when relying upon the steam generators for heat removal. Requiring the EFW pump initiation circuitry to be OPERABLE at ≥ 246°F instead of 250°F is a more restrictive requirement upon unit operation and is consistent with ITS 3.7.5 for the EFW System and the corresponding NUREG Specification.

M26 CTS does not include all Type A and Category 1 post accident monitoring (PAM) instrumentation identified in the plant specific Regulatory Guide 1.97 response and associated NRC Safety Evaluations. ITS 3.3.8 incorporates all Type A and Category 1 PAM Functions consistent with the NUREG. The following PAM Functions, including the associated LCO, Applicability, ACTIONS, Table entries, and Notes, are added:

2. RCS Hot Leg Temperature
4. RCS Pressure (Wide Range)
8. Containment Isolation Valve Position
11. Pressurizer Level
12. Steam Generator Water Level
13. Steam Generator Pressure
15. Upper Surge Tank Level
18. HPI System Flow
19. LPI System Flow
20. Reactor Building Spray Flow

Surveillance Requirements are added (ITS SRs 3.3.8.1 and 3.3.8.3) for PAM Functions 2, 4, 8, 13, 15, and 20. SR 3.3.8.1 is added for PAM Functions 18 and 19. CTS provides comparable Surveillance requirements for pressurizer level (PAM 11)and steam generator water level (PAM 12) indicators (CTS Table 4.1-1, items 26 and 39 respectively). The addition of SR 3.3.8.1 for the Functions designated above is appropriate since a CHANNEL CHECK provides assurance that gross channel failure will be detected and is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The addition of SR 3.3.8.3 for the Functions designated above is appropriate since the CHANNEL CALIBRATION verifies the channel responds to measured parameters within the necessary range and accuracy.

Type A variables are included in the ITS because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accidents (DBAs). Additionally, Category 1 variables are the key variables deemed risk significant because they are needed to: a) determine whether systems important to safety are performing their intended functions; b) provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and c) provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action

necessary to protect the public and to estimate the magnitude of any impending threat. Since these PAM Functions are not in the CTS requirements, their addition represents a more restrictive change.

M27 The CTS applicability for Table 3.5.6-1 instrument 8 is "when RCS temperature is > 300°F". The CTS applicability for other Table 3.5.6-1 PAM instruments is "above hot shutdown." ITS 3.3.8 Applicability for PAM functions is MODES 1, 2, and 3. The CTS applicability of "above hot shutdown" is equivalent to ITS MODES 1 and 2. The CTS applicability of "when RCS temperature is > 300°F" is equivalent to ITS MODES 1 and 2 and part of MODE 3 (Note: transition to MODE 4 is at 250°F). Therefore, the ITS applicability is more restrictive since OPERABILITY is required in MODE 3, but appropriate since these variables are related to the diagnosis and actions required to mitigate accidents that are assumed to occur in these MODES. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low, therefore, PAM instrumentation is not required to be OPERABLE in these MODES. The proposed change is consistent with the NUREG.

Consistent with the more restrictive ITS applicability, Required Action H.2 is added for CTS Table 3.5.6-1 instruments 1 through 7 and 9 requiring the unit be placed in MODE 4 within 18 hours. This is appropriate since it removes the unit from the applicability of the LCO. For CTS Table 3.5.6-1 Instrument 8 (ITS Table 3.3.8-1, Function 17), Action 4 requires the unit to be in hot shutdown within the next 12 hours and below 300°F within the next 24 hours. For this Function, ITS Required Action H.2 requires the unit be in MODE 4 (below 250°F) in 18 hours from the time of entry into the condition versus the total of 36 hours currently allowed. This change is consistent with the NUREG.

M28 CTS SR 3.7.5.1 requires performance of SR 3.7.1.14.1 (EPSL automatic transfer) on a Frequency specified in the applicable SR. CTS SR 3.7.3.1 requires performance of SR 3.7.1.14 on a Frequency specified in the applicable SR. ITS SR 3.3.17.1 and SR 3.3.21.1 require a CHANNEL FUNCTIONAL TEST of the automatic transfer function channels and the Keowee automatic start channels. The CTS test requirement (CTS SR 3.7.1.14) performs a functional verification for the source and Main Feeder Bus voltage sensing, Keowee Emergency start, Loadshed and Transfer-to-Standby, and Retransfer-to-Startup logic of the EPSL System by performing an automatic transfer of the Main Feeder Buses to the Startup Transformer, Standby Buses and retransfer to the Startup Transformers. The method of performing this test is relocated to the Bases for SR 3.3.17.1 and 3.3.21.1 as discussed in DOC LA12. These CTS SRs are captured by the CHANNEL FUNCTIONAL TEST requirements of ITS SR 3.3.17.1 and SR 3.3.21.1. However, the CHANNEL FUNCTIONAL TEST requirements are considered more prescriptive in that they require the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY, including required alarms, interlocks, display, and trip functions. As such, the proposed change is more restrictive on plant operation and consistent with the NUREG.

M29 CTS does not require a calibration of the Wide Range Nuclear instrumentation. ITS SR 3.3.8.3 requires a complete check of the instrument loop and sensor. The addition of this CHANNEL CALIBRATION is appropriate since it verifies the channel responds to measured parameters within the necessary range and accuracy. The proposed change is more restrictive since it is an additional restriction on operation and is consistent with the NUREG.

M30 CTS Table 3.5.1-1 essentially does not have any required actions for ITS 3.3.1 Functions 1.b and 11 when in MODE 3 or lower since Column D only requires the unit be placed in hot shutdown (equivalent to ITS MODE 3). Consistent with the proposed ITS applicability (Refer to DOC M2), required actions are provided that require the unit be placed in a condition outside the applicability of the LCO when the Required Actions cannot be met. Required Action D.1 is added to require the operator to open all the CRD trip breakers in 6 hours. This additional restriction on operation is appropriate since it ensures rod motion is not possible when the required trip channels are inoperable. This change is consistent with the NUREG.

M31 Not used.

M32 Not used.

M33 CTS Table 4.1-1, Column "Check" for Items 54, 57, 58, and 59, currently does not require a check of the containment high range radiation monitor, containment hydrogen monitor, wide range hot leg level, and reactor vessel head level instrument channels. ITS SR 3.3.8.1 is added to require a CHANNEL CHECK of these instrument channels every 12 hours consistent with the NUREG. A CHANNEL CHECK provides reasonable assurance that a gross failure of instrumentation will be identified promptly. The proposed change is more restrictive since it is an additional restriction on operation and is consistent with the NUREG.

M34 CTS 3.7.4 LCO Applicability for Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits is above Cold Shutdown. ITS 3.3.18 LCO Applicability for Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits is MODES 1, 2, 3, 4, 5, and 6 and during movement of irradiated fuel assemblies. ITS MODES 1, 2, 3, and 4 are considered equivalent to the CTS applicability of above Cold Shutdown. ONS design requires that the voltage sensing circuit associated with an AC power source be OPERABLE for the AC power source to be considered OPERABLE. Therefore, since requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies are added (refer to Section 3.8), requirements for EPSL voltage sensing circuits must be added. These additional applicability requirements provide assurance that systems are available to provide adequate coolant inventory makeup, to mitigate a fuel handling accident,and to mitigate the effects of events that can lead to core damage during shutdown and that instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition. LCO Note 2 is added to

specify that only the EPSL voltage sensing circuit(s) associated with required AC power source(s) are required to be OPERABLE. ITS 3.3.18 ACTION C is added to require the affected AC Source to be declared inoperable when the Required Action and associated Completion Time is not met or when two or more channels of a required circuit(s) are inoperable in MODES 5 and 6. ITS 3.3.18 ACTION D is added to require suspending movement of irradiated fuel assemblies when the Required Action and associated Completion Time is not met during movement of irradiated fuel assemblies. Since there are currently no EPSL requirements during cold shutdown and refueling shutdown, the proposed change is more restrictive since it is an additional restriction on operation.

M35 CTS Table 4.1-1 provides no specific CHANNEL CALIBRATION requirement for the Nuclear Overpower High and Low Setpoints. The High and Low Setpoints are calibrated administratively during reactor shutdowns and reactor startups. ITS SR 3.3.1.6 is added for comparable ITS Functions 1.a and 1.b to provide an explicit 18 month CHANNEL CALIBRATION for these functions. This test ensures the channel responds to the measured parameter within the necessary range and accuracy and leaves the channels adjusted to account for instrument drift to ensure that the instrument channel will remain operational between successive tests. A Note to the SR specifically excludes neutron detectors from this CHANNEL CALIBRATION. The addition of SR 3.3.1.6 is considered an appropriate restriction on unit operation since the accident analyses takes credit for these reactor trip functions. The addition of this requirement represents a more restrictive change and is consistent with the NUREG.

M36 CTS 3.4.1 does not allow the reactor to be heated above 250°F unless each emergency feedwater flow (EFW) path has at least one flow indicator operable (CTS 3.4.1.b). ITS 3.3.8, Item 21, Emergency Feedwater Flow, is added to require two channels of EFW flow to be operable in MODES 1, 2 and 3, along with associated Required Actions A, B, and G, the Table entry, and Notes that are applicable for this PAM function. This is consistent with the ONS Regulatory Guide 1.97 Safety Evaluation Report which identifies these indicators as a Category 1 variable and is appropriate to ensure its availability post accident since EFW flow is the primary indication used by the operator to verify that the EFW System is delivering the correct flow to each steam generator. ITS 3.3.8 ACTION A limits the time that one of the two channels can be inoperable to 30 days. If the channel is not restored within 30 days then ITS 3.3.8 ACTION B requires that a written report be submitted to the NRC which identifies the proposed restorative actions and discusses the root cause evaluation. ITS 3.3.8 ACTION G is a pointer to further appropriate action (dependent upon the PAM function) when the Required Action and associated Completion Times for inoperable channels are not met. The addition of these requirements represents a more restrictive change and is consistent with the NUREG.

M37 CTS 3.4.3.b requires the unit be in hot shutdown (equivalent to ITS MODE 3) within 12 hours when no EFW flow indicators in a flow path are

operable and below 250°F (equivalent to ITS MODE 4) within another 12 hours. ITS 3.3.8 Required Action H.2 requires the unit be placed in MODE 4 within 6 hours after reaching MODE 3. The ITS Completion time for Required Action H.2 is more restrictive since the unit must be placed in MODE 4 six hours earlier. The allowed Completion Time is reasonable, based on operating experience, to reach the required unit conditions in an orderly manner and without challenging plant systems. The addition of these requirements represents a more restrictive change and is consistent with the NUREG.

M38 CTS does not require the Manual Keowee Emergency Start function to be OPERABLE below Cold Shutdown. ITS Specification 3.3.22, including appropriate LCO, ACTIONS, and SRs, is added to require one channel of the Manual Keowee Emergency Start function to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies. This is necessitated by the addition of requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies (refer to Section 3.8). Required Action A.1 requires both Keowee Hydro Units to be declared inoperable immediately when the required channel is inoperable. ITS SR 3.3.22.1 requires a CHANNEL FUNCTIONAL TEST of the Keowee manual emergency start function every 12 months. This function is currently tested during the performance of CTS 3.7.1.11. CTS 3.7.1.11 uses the manual start function as a method of initiating the Keowee Hydro Units when verifying they start within 23 seconds and synchronize to the grid. These additional requirements provide assurance that the required AC Sources are available during shutdown. This ensures that systems are available to provide adequate coolant inventory makeup, to mitigate a fuel handling accident, and to mitigate the effects of events that can lead to core damage during shutdown and that instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition. Since there are currently no EPSL requirements during cold shutdown and refueling shutdown, the proposed change is more restrictive since it is an additional restriction on operation.

M39 CTS Table 3.5.1-1 Note (c) allows continued operation above hot shutdown with the required source range instrument channel inoperable when 2 of 4 wide range instrument channels are indicating greater than 4E-4% rated power. ITS 3.3.9 ACTION C, which continues to allow operation above MODE 3 with the required source range instrument channel inoperable, also requires action be initiated to repair the inoperable instrument channel within 1 hour. This more restrictive action is appropriate to ensure future availability. The 1 hour Completion Time is sufficient to initiate the action. This change is consistent with the NUREG.

M40 CTS 3.5.1.1 Applicability for the TSV Closure instrumentation channels is while in the startup mode or when the reactor is in a critical state. ITS 3.3.15 Applicability for the TSV Closure instrumentation channels is in MODES 1, 2, and 3 except when all TSVs are closed. The CTS applicability of "in a critical state" is encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is ≥ 0.99 k_{eff}. CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The expanded applicability is appropriate since during MODE 3 conditions there is significant mass and energy in the RCS and steam generators and the TSV Closure function is needed to stop steam flow to the turbine (to prevent overcooling) following a reactor trip. As such, the addition of applicability in MODE 3 is more restrictive. The more restrictive requirement is consistent with comparable NUREG (3.3.11-1, Function 4) requirements.

TECHNICAL CHANGE-LESS RESTRICTIVE

L1 CTS Table 3.5.1-1 Note (a) allows the minimum of three OPERABLE channels to be maintained during channel testing, calibration, or maintenance by placing one of the four available channels in bypass and one of the four available channels in the tripped condition leaving an effective one out of two trip logic. ITS 3.3.1 Action A also allows this configuration but does not limit its application (i.e., allowed for any reason, not just for channel testing, calibration, or maintenance) to only the CTS reasons. ITS requires the required action be completed within one hour where CTS required this action immediately. The proposed change is acceptable since the RPS can still perform its safety function in this configuration in the presence of a random failure of any single channel. The proposed change is consistent with the NUREG.

L2 CTS 3.5.1.1 requires the RPS functions of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. A critical state is considered encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq 0.99\ k_{eff}$. The CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The applicability for ITS 3.3.1 Function 9, Main Turbine Trip (Hydraulic Fluid Pressure) and Function 10, Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) is less restrictive since these functions are only required to be OPERABLE during MODE 1 (above 30%) and during MODE 1 and MODE 2 (above 2% RTP), respectively. Analyses presented in BAW 1893 show that for operation below these power levels, these trips are not necessary to minimize challenges to the PORV as required by NUREG-0737. Duke Energy has performed a plant specific analysis which concludes that the Oconee RPS System is consistent with the BAW analyses.

As a result of the change in applicability, the CTS Table 3.5.1-1 Column (D) required action of placing the unit in hot shutdown is modified to require reducing power to a level less than the applicability of either function in a time period appropriate for reaching that power level from full power conditions. The proposed change is consistent with the NUREG.

L3 ITS 3.3.2 Action A is added to allow one hour to restore an inoperable manual reactor trip function to OPERABLE status. This is acceptable since the automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour provides a limited time to affect repairs and avoid an unnecessary unit shutdown. This less restrictive change is consistent with the NUREG.

L4 CTS Table 4.1-1 Item 4 requires the power range channel output to the calorimetric coincident with the imbalance output being calibrated to the imbalance condition determined by the incore detector be performed monthly. ITS SR 3.3.1.4, which includes the equivalent requirement, is

modified by a Note that allows this calibration to be delayed as much as 24 hours after THERMAL POWER is ≥ 15% RTP. This SR Note recognizes the difficulty in performing the calibration and the limitations of the calorimetric while operating at very low power levels. Below 15% RTP, ONS calculates heat balance power level based totally upon the primary system parameters. Above 15% RTP, the secondary system parameters are also considered since they are generally more accurate at higher power levels. By allowing the delay in performance of this calibration until RTP is above 15%, a generally more accurate calorimetric (one including secondary system parameters) is available. Also, below about 15% the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. Thus, this allowance is appropriate due to the usable range of the incore nuclear instruments which are required for the performance of this SR.

L5 CTS Table 3.5.1-1 does not include an allowance that allows placing an inoperable ESPS channel in the tripped condition and continuing operation for an indefinite period. ITS 3.3.5 ACTION A is added to allow continued reactor operation for an indefinite period when one of three ESPS channels is inoperable provided the inoperable channel is placed in a tripped condition for actuation within one hour. This less restrictive provision is acceptable since this action leaves the system in a one-out-of-two condition for actuation. Thus, if another channel fails, the ESPS instrumentation can still perform its actuation functions. This less restrictive change is consistent with the NUREG.

L6 CTS does not include an allowance that allows delay of entry into actions when a channel is made inoperable for testing. The Note for ITS SR 3.3.5.2 allows a delay of up to 8 hours in the entry into the associated Condition and Required Action for the performance of this CHANNEL FUNCTIONAL TEST provided the remaining two instrumentation channels are OPERABLE or tripped. This Note provides a relaxation of ACTION requirements which is less restrictive than the application of CTS requirements. This Note provides a reasonable amount of time to perform the required testing while still allowing the channel being tested to remain in an untripped state. Additionally, the design of the ESPS system will not allow complete testing of an instrument channel with the channel in a tripped state. Therefore, placing the channel in a tripped state, as required by ITS 3.3.5 Required Action A.1, prevents the completion of the testing required by SR 3.3.5.2. This change is consistent with the NUREG.

L7 CTS Table 3.5.1-1 requires a unit shutdown in 12 hours when one or more ESPS channels in one or more functions are inoperable. ITS 3.3.6 Required Action A.1 and its associated Completion Time provide a 72 hour time period in which the unit may continue operation, with one or more ESPS Functions having one channel of the manual initiation feature inoperable, prior to entering an ACTION which results in the unit entering MODE 3. This change is made to provide ACTION requirements consistent with the safety function of the system, considering the allowed outage time for the actuated system. Therefore the less

restrictive change is considered appropriate. This change is consistent with the NUREG.

L8 CTS Table 4.1-1 Item 4 requires a calibration of the power range instruments against the incore instruments monthly. This calibration is also required to be performed within some unspecified period of time after each startup if not performed within the previous week. These CTS requirements are replaced by ITS SR 3.3.1.3 and SR 3.3.1.4. The ITS SRs with their specified 31 day Frequencies represents less restrictive requirements in that the calibration is no longer required following each startup if not performed within the previous week. Removal of the required calibration following each startup, is acceptable because deviation between the AXIAL POWER IMBALANCE indicated by the power range instruments and that indicated by the incore instruments generally occurs slowly. The 31 day Frequency is consistent with NUREG.

L9 CTS Table 4.1-1, item 3 requires a heat balance check of the power range channels each shift. ITS SR 3.3.1.2 requires the heat balance check be performed every 24 hours. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds a small fraction of 2% in any 24 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs. The change is consistent with the NUREG

L10 CTS Table 3.5.1-1 Note (i)1 & 2 requires that the power supplied to the CRDMs through the failed CRD be removed within one hour or allows 48 hours to place the breaker in trip if it has diverse features inoperable (undervoltage or shunt trip devices). ITS 3.3.4, Required Action A.2 is added to allow the option of removing power from the CRD trip breaker. This additional allowance is less restrictive in that it provides additional flexibility in dealing with trip breakers with an inoperable diverse trip function. The allowance for removing power from a trip breaker as an alternative to opening the breaker is currently allowed by CTS Table 3.5.1-1 Note (i)1 but is not specifically applicable to the inoperability of the diverse trip function for the trip breakers. The addition of ITS 3.3.4 Required Action A.2 provides consistent ACTION requirements to compensate for inoperable CRD trip breakers whether or not the inoperability is due to failure of a diverse trip function. The Completion Times in ITS remain, as they are in CTS, significantly different for a CRD trip breaker with an inoperable diverse trip function, as opposed to one which is inoperable for any other reason. ITS Required Action B.1 is added to provide the option of tripping the CRD trip breakers that do not have diverse features inoperable. This additional allowance is less restrictive in that it provides additional flexibility in dealing with an inoperable CRD trip breakers. This is appropriate since tripping the inoperable CRD trip breaker has the same effect as removing power to the CRDMs that are powered through the inoperable CRD trip breakers. The proposed changes are consistent with the NUREG.

L11 CTS Table 3.5.1-1 Note j does not include an allowance that allows placing a CONTROL ROD group with an inoperable ETA rely to be placed on a power supply which has OPERABLE ETA relays. ITS 3.3.4 Required Action C.1 is added and provides an alternative to the CTS requirements for inoperable SCR or electronic trip assembly (ETA) relays (CTS Table 3.5.1-1 Note (j)). Required Action C.1 specifically allows for a CONTROL ROD group with an inoperable ETA relay to be placed on a power supply which has OPERABLE ETA relays. This allowance provides new flexibility which is not currently allowed by CTS. Required Action C.1 is an acceptable alternative to opening an inoperable ETA relay because it places the affected CONTROL ROD group on a power supply that ensures the rods are de-energized upon a reactor trip. The change is consistent with the NUREG.

L12 CTS Table 3.5.1-1 Column D and Note (e) require the unit be placed in Hot Shutdown within 24 hours and in Cold Shutdown in the following 72 hours when the minimum ES digital actuation logic channels are not OPERABLE. ITS Required Actions Required Action A.1 and A.2 provide two options. One option is to place the associated component(s) in their ES configuration in one hour. The other is to declare them inoperable (and enter into their associated required actions) within one hour. ITS 3.3.7 Required Action A.1 is equivalent to the automatic actuation logic channel performing its safety function ahead of time. Required Action A.2, which requires entry into the Required Action of the affected supported systems, is appropriate since the net result of the automatic actuation logic failure is inoperability of the supported system. The one hour Completion Time reflects the urgency associated with the inoperability of a actuation logic channel which affects multiple safety system components. ITS 3.3.7 is considered less restrictive since declaring the supported systems inoperable associated with one digital channel would result in only one train of the supported system being declared inoperable. At least a 72 hour Completion Time (Refer to ITS 3.5.2 or 3.6.5 Required Actions and Completion Times) is provided to restore one train of ES actuated components to OPERABLE status prior to requiring a unit shutdown. In addition, where practicable, starting the supported system allows continued operation with no further restrictions. The proposed change is consistent with the NUREG.

L13 CTS 3.5.1.1 requires the source range and wide range instruments to be OPERABLE in a startup mode or in a critical state. CTS Table 3.5.1-1 Note b indirectly provides a qualification to this statement of Applicability. This note provides a relaxation of action requirements when "2 of 4 power range instrument channels are greater than 10% rated power." The Applicability of ITS 3.3.9 and 3.3.10 does not require either the source range instrument channel or the wide range instrument channel to be maintained OPERABLE above MODE 2. This represents a relaxation of requirements, by removing the requirement to take actions in the event that either the required source range instrument channel or the required wide range instrument channel is inoperable, when above 5% RTP (ITS) but less than or equal to 10% rated power, as indicated on the power range instruments (CTS). This is acceptable since the power range

channels provide all assumed reactor protection above 5% RTP. This change is being made to provide clear statements of Applicability for these specifications which are consistent with the requirements of the NUREG.

L14 CTS Table 4.1-1 requires a calorimetric heat balance check and adjustment every shift. ITS SR 3.3.1.2, which requires the verification every 24 hours, is modified by a Note that allows this check be delayed as much as 24 hours after THERMAL POWER is ≥ 15% RTP. The ITS recognizes the difficulty in performing the heat balance check and the limitations of the calorimetric while operating at very low power levels. No specific allowance is provided in the CTS which removes the requirement to perform this calibration when in a critical state at low power levels. Below 15% RTP, ONS calculates heat balance power level based totally upon the primary system parameters. Above 15% RTP, the secondary system parameters are also considered since they are generally more accurate at higher power levels. By allowing the delay in performance of this calibration until RTP is above 15%, a generally more accurate calorimetric (one including secondary system parameters) is available. The proposed change is consistent with the NUREG.

L15 CTS Table 4.1-1 requires a comparison of the out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE every 31 days. ITS SR 3.3.1.3, which provides an equivalent requirement, is modified by a Note that allows a delay in performance of this SR until the unit is above 15% RTP. This allowance is appropriate due to the usable range of the incore nuclear instruments which are required for the performance of this SR. Below about 15% the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. Adoption of this Note provides a specific relaxation of requirements where none existed in CTS. This change is consistent with the NUREG.

L16 CTS 3.4.2, which requires the automatic initiation circuitry associated with loss of main feedwater pumps to be OPERABLE prior to criticality, provides no allowed outage time when one of two loss of main feedwater instrumentation channels are inoperable. ITS 3.3.14, ACTION A is added to allow continued reactor operation for an indefinite period when one of two EFW System loss of main feedwater instrumentation channels in an EFW pump automatic initiation circuit is inoperable provided the inoperable channel is placed in a tripped condition for initiation within one hour. ITS 3.3.14 ACTIONS Note is added to allow separate condition entry for each EFW pump initiation circuit. This allows one hour to place the channel in trip for each function when Condition A is entered. This less restrictive provision is acceptable since this leaves the function in a one-out-of-one logic configuration for initiation versus the normal two-out-of-two logic configuration. This maintains at least equivalent reliability for EFW initiation. EFW is maintained single failure proof by the separate initiation circuits for each the three EFW pumps. This less restrictive change is consistent with NUREG Specification 3.3.11, ACTION A.

L17 CTS 3.7.6 and 3.7.7 both require an inoperable voltage sensing relay to be restored within 72 hours (Required Action A.1). ITS 3.3.19 Required Action A.1 and 3.3.20 Required Action A.1 require the inoperable channel to be placed in trip within 72 hours. This less restrictive change allows operation to continue indefinitely when the channel is placed in trip and continues to allow 72 hours to restore an inoperable channel that cannot be placed in trip. The actuation logic for DGVP is two-out-of-three. Placing the inoperable channel in the tripped condition fulfills the function of the channel (and places the function in a one-out-of-two configuration). Indefinite operation in this configuration is acceptable since the degraded grid voltage function is capable of performing its function in the presence of a single failure. This change is consistent with comparable NUREG 3.3.8 requirements.

L18 CTS 3.5.7 Applicability for the Main Steam Line Break and Feedwater Isolation Circuity is when main steam header pressure is greater than 700 psig. ITS 3.3.11, 12, & 13 Applicability is MODES 1 and 2, and MODE 3 with main steam header pressure greater than 700 psig except when all MFCVs and SFCVs are closed. The exception of "when all MFCVs and SFCVs are closed" is a less restrictive change and is consistent with comparable NUREG requirements (Table 3.3.11-1, Note d). The exception is appropriate since the MFCVs and SFCVs are already performing their safety function when they are closed.

Required Action B.2.2 of ITS 3.3.11, 12 and 13 is added to provide the option of closing the MFCVs and SFCVs in lieu of reducing main steam header pressure to less than 700 psig. This optional allowance is consistent with the applicability since closure of the MFCVs and SFCVs removes the unit from the Applicability of the LCO.

L19 CTS Table 3.5.1-1 requires a unit shutdown within 24 hours when one or more turbine stop valve closure channels is inoperable. ITS 3.3.15 ACTION A is added to allow one hour to declare the associated TSVs inoperable. The additional hour allowed to restore the instrumentation channel(s) prior to requiring further action is consistent with similar NUREG Required Actions that require supported equipment to be declared inoperable (e.g., NUREG Specification 3.3.7, Required Action A.2). The one hour Completion Time is considered sufficient to correct minor problems. Even with the 1 hour, the unit gets to subcriticality sooner (13 hours) than that time allowed by CTS Table 3.5.1-1, Column D for Item 16 (24 hours).

L20 CTS Table 4.1-1 calibration requirements for RPS functions that receive input from neutron detectors do not specifically exclude the detectors from the calibration of that function. ITS SR 3.3.1.6, which provides comparable CHANNEL CALIBRATION requirements for RPS functions, includes a note that specifically excludes the neutron detectors from the CHANNEL CALIBRATION. This exclusion is appropriate because of the passive design of the detectors, the extreme difficulty in both accessing the detectors and in generating an appropriate input signal to the detectors, the fact that no specific adjustments can be made to the

detectors, and the principles of detector operation that ensure a virtually instantaneous response. The proposed change is consistent with the NUREG.

L21 CTS Table 3.5.6-1 Action 3 for the Reactor Vessel Head Level and the Reactor Vessel Level (ITS 3.3.8 PAM #3 and #5) allows, if repairs are feasible, 7 days for restoration of a single inoperable instrument channel when one or both instrument channels are inoperable. Operation may continue with one inoperable channel, provided a report is submitted within the next 30 days outlining the cause of the inoperability and the plans and schedule for restoring the channel to OPERABLE status. When both are inoperable, if at least one instrument channel is not restored, the unit is then required to be in hot shutdown within 12 hours. ITS ACTION A allows 30 days for restoration of a single channel, and ITS ACTION C allows 7 days for restoration of one of two inoperable instrument channels. ITS ACTIONs B and I then require a Special Report. Therefore, the proposed Required Action I.1 is less restrictive since a unit shutdown is not required. Required Action I.1 is appropriate in lieu of a shutdown requirement since both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling capability and both the subcooling monitoring monitors and core exit thermocouples provide an alternate means of monitoring for this purpose. This change is consistent with the NUREG.

L22 The CTS Actions (Table 3.5.6-1, Action 1) for the Containment Pressure - High Range (PAM #7) Function allow 7 days for restoration of an inoperable instrument channel and 48 hours for restoration of an inoperable channel when both are inoperable. When either required action is not met, the unit must be placed in hot shutdown within the next 12 hours. ITS 3.3.8, ACTION A allows 30 days for restoration of a single channel. If not restored, then a Special Report is required by Required Action B.1. This is less restrictive since a unit shutdown is not required. ITS Required Action B.1 is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. ITS ACTION C allows an additional 5 days for restoration of a single channel when both channels are inoperable. The additional time allowed to restore at least one channel allowed by Action C is considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. These less restrictive changes are consistent with the NUREG.

L23 The CTS Actions (Table 3.5.1-1, Action 2) for Containment Water Level (PAM #6), Containment High-Range Radiation (PAM #9), Containment Hydrogen (PAM #10), and the Core Exit Thermocouple (PAM #16) Functions allow 30 days for restoration of an inoperable instrument channel and 48 hours for restoration of an inoperable channel when both are inoperable. When either required action is not met, the unit must be placed in hot shutdown within the next 12 hours. ITS 3.3.8, ACTION A allows 30 days

for restoration of a single channel and then a Special Report is required by Required Action B.1. ITS Required Action B.1 is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. This is less restrictive since a unit shutdown is not required. ITS ACTION C for the Containment Water Level, Containment High-Range Radiation, and the Core Exit Thermocouple Functions allows an additional 5 days for restoration of a single channel when both channels are inoperable. ITS ACTION D for the Containment Hydrogen Concentration Function allows an additional 24 hours. The additional time allowed to restore at least one channel allowed by ITS Actions C and D are considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. ITS 3.3.8, Action I is added for ITS Table 3.3.8-1, Function 9 (Containment High-Range Radiation), to allow a Special Report in place of the CTS requirement for shutdown. This is acceptable since alternate means are available to monitor this variable. These changes are consistent with the NUREG.

L24 The CTS Actions (Table 3.5.1-1, Action 4) for the Subcooling Monitor function (PAM #17) Function allows 30 days for restoration of an inoperable instrument channel and 48 hours for restoration of an inoperable channel when both are inoperable. When either required action is not met, the unit must be placed in hot shutdown within the next 12 hours. ITS 3.3.8, ACTION A allows 30 days for restoration of a single channel and then a Special Report is required by ITS Required Action B.1. ITS Required Action B.1 is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. This is less restrictive since a unit shutdown is not required. ITS ACTION C allows an additional 5 days for restoration of a single channel when both channels are inoperable. The additional time allowed to restore at least one channel allowed by ITS Action C is considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. These changes are consistent with the NUREG.

L25 Not used.

L26 CTS Table 4.1-1 requires a CHANNEL CHECK of items 5, 26, 30, and 39 either shiftly or weekly. ITS SR 3.3.8.1 requires a CHANNEL CHECK of PAM instrument channels for each required channel that is normally energized every 31 days. The Frequency is based on operating experience that demonstrates channel failure is rare, and on the use of less formal but more frequent checks of channels during normal operational use of the displays associated with the required channels. This less restrictive change is consistent with the NUREG.

L27 CTS Table 4.1-1, items 54 and 57, requires a monthly functional test of the containment high range radiation monitor and containment hydrogen monitor instrument channels. This monthly functional test is not included in ITS. Such a test is typically required when the instrumentation provides a safety related automatic actuation function. This instrument channel provides information only, and as such, a CHANNEL FUNCTIONAL TEST is not appropriate, nor required. This change is also consistent with the NUREG.

L28 CTS Table 3.5.1-1, Column D requires the unit be placed in hot shutdown within 12 hours when less than two source range channels are OPERABLE and rated power is $\leq$ 10% as shown on the power range channels and $\leq 4 \times 10^{-4}$ % rated power as shown on the wide range channels. Comparable ITS Required Actions do not require the unit be placed in hot shutdown. Therefore, the proposed ITS Required Actions are less restrictive in this aspect. However, other more appropriate required actions are added to replace the CTS required action (Refer to DOC M13). This change is also consistent with the NUREG.

L29 Not used.

L30 CTS 3.5.1.5 requires the overlap between the wide range and the source range instrumentation to be checked during startup. Proposed ITS SR 3.3.10.3 requires the overlap to be verified every startup if not performed within the previous 7 days. The ITS allows the test to be omitted if performed within the previous 7 days. This is based on industry operating experience which shows the instrument overlap does not change appreciably within this test interval. The proposed change is consistent with the NUREG.

L31 CTS Table 3.5.1-1, Column D requires the unit be placed in hot shutdown within 12 hours when less than two wide range channels are OPERABLE. ITS 3.3.10, Required Action A.1 only requires that power be reduced to $< 4 \times 10^{-4}$% RTP when one channel is inoperable. The proposed change is less restrictive since the CTS defines Hot Shutdown as the reactor having a K_{eff} of ≤ 0.99 and the reactor could have a K_{eff} of > 0.99 with power reduced below 4×10^{-4}% RTP as allowed by Required Action A.1. The proposed change is consistent with the NUREG.

L32 CTS Table 4.1-1, Column "Test," Items 5 and 6 require a functional test be performed on the source range and wide range channels prior to startup. This requirement is not retained in the ITS. Consistent with the NUREG, a CHANNEL CALIBRATION of the source range and wide range instruments is added (Refer to DOC M14). Because the calibration by definition encompasses the functional test, performance of the calibrations will ensure that testing is consistent with CTS requirements. The frequency of this testing is now based strictly on the time since its last performance and not dependent upon whether or not the unit is in startup. This change is acceptable, based on operating experience which demonstrates the source and wide range instruments are highly reliable.

L33 CTS 3.7.5.1 requires performance of SR 3.7.1.11 (Keowee emergency start) and SR 3.7.1.14 (EPSL automatic transfer). SR 3.7.1.11 verifies that each Keowee Hydro Unit (KHU) can emergency start from each control room, attain rated speed and voltage within 23 seconds of an emergency start initiate, and be synchronized to the grid and loaded. The test is performed by manually starting one KHU from the Unit 1 and 2 Control Room and the other KHU from the Unit 3 Control Room. The accident analyses do not take credit for a manual Keowee start during operation above Cold Shutdown. Therefore, the requirement to test this function during operation above Cold Shutdown is not retained. This function is required to be OPERABLE during MODES 5 and 6 and during movement of irradiated fuel assemblies by ITS 3.3.22, "EPSL Manual Keowee Emergency Start Function."

L34 CTS 3.5.1.1 Applicability for the TSV Closure instrumentation channels is while in the startup mode or when the reactor is in a critical state. This is considered encompassed by ITS MODES 1 and 2. ITS 3.3.15 Applicability is in MODES 1, 2, and 3 except when all TSVs are closed. The exception of "when all TSVs are closed" is a less restrictive change and is consistent with comparable NUREG requirements (Table 3.3.11-1, Note c). The exception is appropriate since the TSVs are already performing their safety function when they are closed.

L35 CTS 3.8.10 requires the radiation monitor associated with the purge system valve isolation to be tested and verified OPERABLE immediately prior to refueling operations. CTS Table 4.1-2, Item 4, requires this functional test be performed "Prior to Refueling." ITS 3.3.16 Applicability is during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment. ITS SR 3.3.16.2 requires the testing be performed once each refueling outage prior to CORE ALTERATIONS or beginning movement of irradiated fuel assemblies within containment. Permitting the specified testing to be conducted prior to beginning movement of irradiated fuel assemblies within containment in lieu of immediately prior to refueling operations is a less restrictive requirement upon unit operation (and is more stringent than the NUREG). Requiring performance of SR 3.3.16.2 once each refueling outage prior to CORE ALTERATIONS or prior to beginning movement of irradiated fuel assemblies within containment represents a reasonable relaxation of the CTS surveillance frequency. This continues to ensure that this function is verified prior to irradiated fuel assembly handling within containment.

L36 CTS Table 3.5.1-1, Column D requires the unit to be in hot shutdown within 24 hours when one or more TSV Closure Instrumentation channels is inoperable and Note (e) to the Table requires the unit be placed in Cold Shutdown within the following 72 hours if the minimum conditions are not met. ITS 3.3.15 ACTION A is added to require the TSVs to be declared inoperable within 1 hour (also, see DOC L19). ITS 3.7.2, Turbine Stop Valves, then dictates the required action for inoperable TSVs. With one or more TSVs inoperable in MODE 1, Required Action A.1 requires the TSVs be restored to OPERABLE status within 8 hours or Required Action B.1

requires the unit be in MODE 2 in 6 hours. Therefore, this portion of ITS is more restrictive since the unit must be in MODE 2 within 15 hours of an inoperable TSV Closure instrumentation channel where CTS required the unit be in hot shutdown (equivalent to ITS MODE 3) within 24 hours. ITS 3.7.2 Action C allows 8 additional hours to close an inoperable TSV when in MODE 2 or 3 (total of 23 hours). In addition, if it were not closed, then an additional 12 hours (on top of the eight hours) is allowed to place the unit in MODE 3 and 18 hours to place the unit in MODE 4. This results in allowing a total of 35 hours to be in MODE 3 and 41 hours to be in MODE 4 from initial discovery of it being inoperable in MODE 1. This compares to the CTS time allowed to place the unit in Hot Shutdown (MODE 3) of 24 hours. Therefore, an additional 11 hours is allowed to place the unit in MODE 3. The additional time is reasonable considering the low probability of an accident occurring during this time period that would require closure of the TSVs. The more restrictive aspects of this change are addressed in DOC M22. The proposed less restrictive ITS Shutdown Times requirements are consistent with ITS 3.7.2, which is consistent with the NUREG.

L37 CTS 3.4.3.b requires a flow path with no OPERABLE emergency feedwater flow indicators to be restored to OPERABLE status within 72 hours. ITS 3.3.8 Required Action C.1 allows 7 days to restore an inoperable flow indicator when both are inoperable. Required Action C.1 allows an additional 4 days for restoration of a single channel when no channels are OPERABLE. The additional time to restore at least one channel allowed by Required Action C.1 is considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. This less restrictive change is consistent with the NUREG.

LESS RESTRICTIVE-REMOVAL OF DETAILS

LA1 CTS Table 2.3-1, Notes (1) and (2), which provide information regarding how the shutdown bypass setpoints are controlled or set, are relocated to the Bases of ITS 3.3.1. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for system OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA2 CTS Table 4.1-1, Item 2, Remarks column specifies that the functional test for this channel shall independently confirm the operability of the shunt trip device and the undervoltage device. These requirements are relocated to ITS SR 3.3.4.1 Bases. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirements for OPERABILITY of the trip breaker function. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA3 CTS Table 3.5.1-1, Columns (A) and (B) provide the total channels available for each function and the number of channels required to trip each function. This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications.

LA4 Note h to Table 3.5.1-1 provides details on the number of RCP monitor channels required OPERABLE for the RCP monitor logic to be considered OPERABLE. CTS states that for OPERABILITY to be met either all RCP monitor channels must be OPERABLE or 3 OPERABLE with the remaining channel in the tripped state. These requirements are relocated to UFSAR Chapter 16. These details are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirements for the Reactor Coolant Pump to Power Function. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

LA5 CTS Table 3.5.6-1, Column A lists the number of required channels in terms of "2 of 2" or "2 of 4." ITS Table 3.3.8-1 lists only the required number (i.e., "2"). The total number of channels available is moved to the Bases for ITS 3.3.8. CTS Table 3.5.6-1 provides an equipment number along with the description of the PAM instrument for Items 1 through 5. These equipment numbers are repeated in CTS Table 4.1-1. This detail is moved to the Bases for ITS 3.3.8. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. This change is consistent with the NUREG.

LA6 CTS 3.5.3 and CTS Table 4.1-1 Remarks column provide details on equipment started by an ESPS signal. This information has been moved to the Bases for ITS 3.3.5, 3.3.6, and 3.3.7. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications.

LA7 CTS 3.4.2 refers to auto-initiation circuitry associated with loss of main feedwater pumps as sensed by hydraulic oil pressure. This

description of the method of sensing loss of main feedwater pumps is relocated to the Bases for ITS 3.3.14. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for the auto-initiation circuitry. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA8 CTS 3.5.1.3 and 3.5.1.4 specify details on how channels are bypassed and requires controls on the bypass key. This information has been moved to the Bases for ITS 3.3.1. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications.

LA9 CTS Table 3.5.1-1, Note (a) provides information regarding how the minimum of three operable channels may be maintained, are relocated to the Bases of ITS 3.3.1. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for system OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA10 CTS Table 3.5.1-1, Functional Unit 19 includes specific SCR (ETA) control relay letter designators "E" and "F" in the description of the function. These designators are also included in Table 4.1-1 for Item 2. This information is relocated to the Bases of ITS 3.3.4. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for system OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is

unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA11 CTS Table 3.5.6-1 Note b states that operable subcooling margin monitors must consist of one direct indication for 1 of 2 RCS hot legs and one direct indication for the core; or one direct indication for each RCS hot leg. This information is relocated to the Bases of ITS 3.3.8. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for system OPERABILITY. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA12 CTS SR 3.7.1.14 requires performing an automatic transfer of the Main Feeder Buses to the Startup Transformer, Standby Buses and retransfer to the Startup Transformers to verify the source and Main Feeder Bus voltage sensing, Keowee Emergency start, Loadshed and Transfer-to-Standby, and Retransfer-to-Startup logic of the EPSL System is functional. The method of performance of this surveillance is relocated to the Bases for ITS SR 3.3.17.1 and SR 3.3.21.1. The ITS retains the requirement to perform the CHANNEL FUNCTIONAL TEST which ensure OPERABILITY of the automatic transfer function. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA13 CTS 3.7.5.1 requires performance of SR 3.7.1.11 (Keowee emergency start) and SR 3.7.1.14 (EPSL automatic transfer). SR 3.7.1.11 verifies that each Keowee Hydro Unit (KHU) can emergency start from each control room, attain rated speed and voltage within 23 seconds of an emergency start initiate, and be synchronized to the grid and loaded. The test is performed by manually starting one KHU from the Unit 1 and 2 Control Room and the other KHU form the Unit 3 Control Room. This manual Keowee

start is a method of performing the test not a requirement for meeting the test. This detail is relocated to the Bases for ITS SR 3.8.1.11. The ITS retains the requirement to test the automatic start function for the KHUs. The accident analyses do not take credit for a manual Keowee start during operation above Cold Shutdown. This detail is not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for testing the automatic start function. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable. Changes to the Bases are controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA14 CTS 3.8.10 provides an associated equipment number when referring to the radiation monitor associated with purge valve initiation. This detail is moved to the Bases for ITS 3.3.16. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. This change is consistent with the NUREG.

RELOCATED SPECIFICATIONS

R1 CTS Table 4.1-1 requirements associated with:

Item 22, pressurizer temperature indicators,
Item 25a (and CTS 3.3.3), core flood pressure indicators,
Item 25b (and CTS 3.3.3), core flood tank level indicators,
Item 27, letdown storage tank level indicators,
Item 31, Boric Acid Mix Tank Level and Temperature,
Item 32, CBAST Level and Temperature,
Item 33, Containment Temperature,
Item 35, Emergency Plant Radiation Instruments,
Item 36, Environmental Monitors,
Item 38, Reactor Building Emergency Sump Level,
Item 40, Turbine Overspeed Trip and
Item 50, PORV and Safety Valve Position indicators;

are relocated to UFSAR Chapter 16. These requirements are not retained in the ITS because they have been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the Oconee Nuclear Station Unit 1, 2, and 3 Technical Specifications."

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.3 - INSTRUMENTATION

ATTACHMENT 4

NO SIGNIFICANT HAZARDS CONSIDERATIONS

ADMINISTRATIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve reformatting, renumbering, and rewording of Technical Specifications. These changes, since they do not involve technical changes to the Technical Specifications, are administrative.

This type of change is connected with the movement of requirements within the current requirements, or with the modification of wording which does not affect the technical content of the current Technical Specifications. These changes will also include nontechnical modifications of requirements to conform to the Writer's Guide or provide consistency with the Improved Standard Technical Specifications in NUREG-1430. Administrative changes are not intended to add, delete, or relocate any technical requirements of the current Technical Specifications.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specification. These modifications involve no technical changes to the existing Technical Specifications. The majority of changes were done in order to be consistent with NUREG-1430. During the development of NUREG-1430, certain wording preferences or English language conventions were adopted. The changes are administrative in nature and do not impact initiators of analyzed events. They also do not impact the assumed mitigation of accidents or transient events. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes do not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The changes will not impose any new or different requirements or eliminate any existing requirements. Therefore, the changes do not create the

possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in margin of safety?**

The proposed changes involve reformatting, renumbering, and rewording of the existing Technical Specifications. The changes are administrative in nature and will not involve any technical changes. The changes will not reduce a margin of safety because it has no impact on any safety analysis assumptions. Also, since these changes are administrative in nature, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

MORE RESTRICTIVE CHANGES

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve adding more restrictive requirements to the existing Technical Specifications by either making current requirements more stringent or by adding new requirements which currently do not exist.

These changes may include additional commitments that decrease allowed outage time, increase frequency of surveillance, impose additional surveillance, increase the scope of a specification to include additional plant equipment, increase the applicability of a specification, or provide additional actions. These changes are generally made to conform with the NUREG-1430.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event. If anything the new requirements may decrease the probability or consequences of an analyzed event by incorporating the more restrictive changes. The changes do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. The changes do not alter the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The changes do impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis.

Therefore, the changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in margin of safety?**

The proposed changes provide more stringent requirements than previously existed in the Technical Specifications. Adding more restrictive requirements either increases or has no impact on the margin of safety. The changes, by definition, provide additional restrictions to enhance plant safety. The changes maintain requirements within the safety analyses and licensing basis. As such, no question of safety is involved. Therefore, the changes do not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGES - REMOVAL OF DETAILS

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve moving details (engineering, procedural, etc.) out of the Technical Specifications and into a licensee controlled document. This information may be moved to the ITS Bases, UFSAR, plant procedures or other programs controlled by the licensee. The removal of this information is considered to be less restrictive because it is no longer controlled by the Technical Specification change process. Typically, the information moved is descriptive in nature and its removal conforms with NUREG-1430 for format and content.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed changes move details from the Technical Specifications to a licensee controlled document. The changes do not result in any hardware or operating procedure changes. The details being removed from the Technical Specifications are not assumed to be an initiator of any analyzed event. The licensee controlled documents containing the removed Technical Specification details are maintained using the provisions of 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs. Since changes to a licensee controlled document are evaluated per 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated is involved. Therefore, the changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed changes move detail from the Technical Specifications to a licensee controlled document. The changes will not alter the plant configuration (no new or different type of equipment will be installed) or make changes in methods governing normal plant operation. The changes will not impose different requirements, and adequate control of information will be maintained. The changes will not alter assumptions made in the safety analysis and licensing basis. Therefore, the changes will not create the

possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The proposed changes move detail from Technical Specifications to a licensee controlled document. The changes do not reduce the margin of safety since the location of details has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specification to a licensee controlled document are the same as the existing Technical Specification. Future changes to this licensee controlled document will be evaluated per the requirements of 10 CFR 50.59, 10 CFR 50.54(a), 10 CFR 50.55(a), or other established review and control programs.

RELOCATED SPECIFICATIONS

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." Some of the proposed changes involve relocating existing Technical Specification Requirements and Surveillances to licensee controlled documents.

Duke Energy has evaluated the current Technical Specifications using the criteria set forth in 10 CFR 50.36. Specifications identified by this evaluation that did not meet the retention requirements specified in the regulation are not included in the Improved Technical Specifications (ITS) submittal. These specifications have been relocated from the current Technical Specifications to licensee controlled programs, including the UFSAR and Selected Licensee Commitments Manual.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated these proposed Technical Specification changes and determined they do not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components, or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the Oconee Nuclear Station Technical Specifications. The requirements are relocated from the Technical Specifications to licensee controlled documents which will be maintained pursuant to 10 CFR 50.59 thereby reducing the level of regulatory control. The level of regulatory control has no impact on the probability or consequences of an accident previously evaluated. Therefore, the change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change relocates requirements and surveillances for structures, systems, components, or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the Oconee Nuclear Station Technical Specifications. The change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change will not impose different requirements, and adequate control of information will be maintained. This change will not alter assumptions made in the

safety analysis and licensing basis. Therefore, the change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change relocates requirements and surveillances for structures, systems, components, or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the Oconee Nuclear Station Technical Specifications. The change will not reduce a margin of safety since the location of a requirement has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component, or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, there will be no reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L1

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1 Note (a) allows the minimum of three OPERABLE channels to be maintained during channel testing, calibration, or maintenance by placing one of the four available channels in bypass and one of the four available channels in the tripped condition leaving an effective one out of two trip logic. ITS 3.3.1 Action A also allows this configuration but does not limit its application (i.e., allowed for any reason, not just for channel testing, calibration, or maintenance) to only the CTS reasons. ITS requires the required action be completed within one hour where CTS required this action immediately. The proposed change is acceptable since the RPS can still perform its safety function in this configuration in the presence of a random failure of any single channel. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change allows indefinite continued operation with one required channel inoperable provided one of the four available channels is placed in a tripped condition within one hour. This change does not result in any hardware changes. The RPS is not considered as the initiator of any previously analyzed accident. As such, the probability of an accident is independent of the status of the RPS. Since the CTS allows continued operation in this MODE, during channel testing, calibration, and maintenance, any increase in the probability of a spurious trip is not considered significant. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that originally considered. The consequences are not changed since the RPS functions the same, regardless of the reason for placing the channel in trip. Therefore, the change does not significantly increase the probability or consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required trip functions. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   This change to the RPS requirements does not involve a change in setpoints and cannot affect any margin of safety associated with the response to a design basis accident. The RPS is currently allowed to operate with the RPS trip functions in conditions which are not single failure proof to prevent a spurious trip during channel testing, calibration, or maintenance. Therefore, this change to allow the RPS trip functions to operate indefinitely with one required RPS trip channel inoperable provided one channel is placed in the tripped condition within one hour, regardless of the reason, is not considered to involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L2

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS 3.5.1.1 requires the RPS functions of Table 3.5.1-1 to be OPERABLE when the reactor is in a startup mode or in a critical state. A critical state is considered encompassed by ITS MODES 1 and 2, which are defined as MODES where the reactivity condition is $\geq 0.99\ k_{eff}$. The CTS defines the startup mode to be when the shutdown margin is reduced with the intent of going critical. This is considered equivalent to ITS MODE 2 as described in the associated DOCs for Section 1.0. The applicability for ITS 3.3.1 Function 9, Main Turbine Trip (Hydraulic Fluid Pressure) and Function 10, Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) is less restrictive since these functions are only required to be OPERABLE during MODE 1 (above 30%) and during MODE 1 and MODE 2 (above 2% RTP), respectively. Analyses presented in BAW 1893 show that for operation below these power levels, these trips are not necessary to minimize challenges to the PORV as required by NUREG-0737. Duke Energy has performed a plant specific analysis which concludes that the Oconee RPS System is consistent with the BAW analyses.

As a result of the change in applicability, the CTS Table 3.5.1-1 Column (D) required action of placing the unit in hot shutdown is modified to require reducing power to a level less than the applicability of either function in a time period appropriate for reaching that power level from full power conditions. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The Applicability for the Loss of Main Feedwater Pumps reactor trip Function and the Main Turbine Trip reactor trip Function is changed from "above hot shutdown" to $\geq$ 2% RTP and $\geq$ 30% RTP respectively. Similarly, the Required Actions have been revised to require only that the condition of Applicability be exited. This change in Applicability and Required Actions for these

functions does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original Applicability since these trip functions are bypassed during the Conditions which will be omitted from the revised Applicability. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required trip functions. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The Loss of Main Feedwater Pumps reactor trip Function and the Main Turbine Trip reactor trip Function provide anticipatory trips under certain operating conditions. In the conditions to be excluded from the Applicability, the trip functions are bypassed and provide no input to the safety analysis. Therefore, the changes do not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L3

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> ITS 3.3.2 Action A is added to allow one hour to restore an inoperable manual reactor trip function to OPERABLE status. This is acceptable since the automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour provides a limited time to affect repairs and avoid an unnecessary unit shutdown. This less restrictive change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The manual reactor trip function is not an initiator of analyzed events. The proposed change allows one hour to restore an inoperable manual reactor trip function to OPERABLE status. The probability of an event occurring during the additional one hour allowed by the ITS actions, is no greater than the probability of an event occurring during the CTS Actions (12 hours allowed to bring the unit to hot shutdown). In addition, the consequences of an event occurring during the additional one hour allowed by the proposed change are no different than the consequences of the event occurring during the 12 hours allowed by CTS to bring the reactor to hot shutdown. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt

restoration of compliance with the limiting condition for operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed addition of a short restoration time has been determined appropriate based on a combination of the time required to perform the action, the alternate manual trip methods available, and the automatic trip functions available. Therefore, the addition of a short one hour restoration time prior to requiring the initiation of a unit shutdown involves no significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L4

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-1 Item 4 requires the power range channel output to the calorimetric coincident with the imbalance output being calibrated to the imbalance condition determined by the incore detector be performed monthly. ITS SR 3.3.1.4, which includes the equivalent requirement, is modified by a Note that allows this calibration to be delayed as much as 24 hours after THERMAL POWER is ≥ 15% RTP. This SR Note recognizes the difficulty in performing the calibration and the limitations of the calorimetric while operating at very low power levels. Below 15% RTP, ONS calculates heat balance power level based totally upon the primary system parameters. Above 15% RTP, the secondary system parameters are also considered since they are generally more accurate at higher power levels. By allowing the delay in performance of this calibration until RTP is above 15%, a generally more accurate calorimetric (one including secondary system parameters) is available. Also, below about 15% the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. Thus, this allowance is appropriate due to the usable range of the incore nuclear instruments which are required for the performance of this SR.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   A Note is included which allows deferral of calibrating the power range channels (excore) to the incore channels at low power levels. This change in applicability for this Surveillance does not result in any hardware changes. The power range monitors are not considered as initiators for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. Performance of this Surveillance at low power levels generally provides less accurate results than at higher power levels. Since the results of the Surveillance are typically small adjustments, the change

which allows nonperformance during low power does not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure appropriate availability for the instrument channels considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The power range instrument channels provide no identifiable margin of safety at low power since their calibration to the calorimetric heat balance and incore power imbalance does not provide accurate results. In the conditions to be excluded from the Surveillance, the power range instrumentation is available, but calibration is recognized as uncertain. Therefore, the change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L5

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1 does not include an allowance that allows placing an inoperable ESPS channel in the tripped condition and continuing operation for an indefinite period. ITS 3.3.5 ACTION A is added to allow continued reactor operation for an indefinite period when one of three ESPS channels is inoperable provided the inoperable channel is placed in a tripped condition for actuation within one hour. This less restrictive provision is acceptable since this action leaves the system in a one-out-of-two condition for actuation. Thus, if another channel fails, the ESPS instrumentation can still perform its actuation functions. This less restrictive change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change allows indefinite continued operation with one required channel inoperable provided it is placed in a tripped condition within one hour. This action leaves the system in a one-out-of-two condition for actuation. Thus, if another channel were to fail, the ESPS instrumentation can still perform its actuation functions. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the ESPS does not change (and therefore any initiation scenarios are not changed). Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that originally considered. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required ESPS actuation functions. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   This change to the ESPS requirements does not involve a change in setpoints and cannot affect any margin of safety associated with the response to a design basis accident. The proposed change does not prevent the ESPS instrumentation from performing their actuation functions since the action places the ESPS instrumentation in a one-out-of-two condition for actuation versus the normal two-out-of-three logic. Thus, if another channel were to fail, the ESPS instrumentation could still perform its actuation functions. Therefore, this change to allow the ESPS actuation functions to operate indefinitely with one required ESPS actuation channel inoperable provided the channel is placed in the tripped condition within one hour, is not considered to involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L6

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS does not include an allowance that allows delay of entry into actions when a channel is made inoperable for testing. The Note for ITS SR 3.3.5.2 allows a delay of up to 8 hours in the entry into the associated Condition and Required Action for the performance of this CHANNEL FUNCTIONAL TEST provided the remaining two instrumentation channels are OPERABLE or tripped. This Note provides a relaxation of ACTION requirements which is less restrictive than the application of CTS requirements. This Note provides a reasonable amount of time to perform the required testing while still allowing the channel being tested to remain in an untripped state. Additionally, the design of the ESPS system will not allow complete testing of an instrument channel with the channel in a tripped state. Therefore, placing the channel in a tripped state, as required by ITS 3.3.5 Required Action A.1, prevents the completion of the testing required by SR 3.3.5.2. This change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The extension of the Completion Time for the Required Action, provided by this SR Note, does not result in any hardware changes. The Completion Time for performance of Required Actions, even when extended by the allowance of this Note, does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Also, an extension of the Completion Time provides additional opportunity to perform required testing and avoid the increased potential for a transient during the shutdown process. Further, the extension of the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the previous evaluation of accidents.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed extension to the Completion Time has been determined appropriate based on a combination of the time required to perform the required testing, the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the extension of the Completion Time involves no significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L7

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS Table 3.5.1-1 requires a unit shutdown in 12 hours when one or more ESPS channels in one or more functions are inoperable. ITS 3.3.6 Required Action A.1 and its associated Completion Time provide a 72 hour time period in which the unit may continue operation, with one or more ESPS Functions having one channel of the manual initiation feature inoperable, prior to entering an ACTION which results in the unit entering MODE 3. This change is made to provide ACTION requirements consistent with the safety function of the system, considering the allowed outage time for the actuated system. Therefore the less restrictive change is considered appropriate. This change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance of Required Actions does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the previous evaluation of accidents.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short Completion Time involves no significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L8

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-1 Item 4 requires a calibration of the power range instruments against the incore instruments monthly. This calibration is also required to be performed within some unspecified period of time after each startup if not performed within the previous week. These CTS requirements are replaced by ITS SR 3.3.1.3 and SR 3.3.1.4. The ITS SRs with their specified 31 day Frequencies represents less restrictive requirements in that the calibration is no longer required following each startup if not performed within the previous week. Removal of the required calibration following each startup, is acceptable because deviation between the AXIAL POWER IMBALANCE indicated by the power range instruments and that indicated by the incore instruments generally occurs slowly. The 31 day Frequency is consistent with NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change does not result in any hardware changes or changes in operating methods. The change removes an unnecessary additional performance of a surveillance which has been performed within its normal monthly Frequency. Not performing the surveillance at the startup would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure adequate surveillance is performed to identify any degradation of the power range instrumentation channel. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for a power range instrument channel is based on availability and capability of the instrument to perform its safety function. Since the monthly Frequency is adequate to confirm the availability and capability, the removal of an additional confirmatory check of the instrumentation does not impact that availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L9

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-1, item 3 requires a heat balance check of the power range channels each shift. ITS SR 3.3.1.2 requires the heat balance check be performed every 24 hours. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds a small fraction of 2% in any 24 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs. The change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change replaces the 12 hour interval for performance of a heat balance calibration of the power range instruments with a 24 hour interval. This change allows less frequent performances of this Surveillance Requirement. A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and since the proposed Frequency has been determined to be adequate to demonstrate reliable operation of the equipment. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because, a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

Changes in the surveilled parameter occur relatively slow during the proposed intervals, and the proposed Frequency is sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L10

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1 Note (i)1 & 2 requires that the power supplied to the CRDMs through the failed CRD be removed within one hour or allows 48 hours to place the breaker in trip if it has diverse features inoperable (undervoltage or shunt trip devices). ITS 3.3.4, Required Action A.2 is added to allow the option of removing power from the CRD trip breaker. This additional allowance is less restrictive in that it provides additional flexibility in dealing with trip breakers with an inoperable diverse trip function. The allowance for removing power from a trip breaker as an alternative to opening the breaker is currently allowed by CTS Table 3.5.1-1 Note (i)1 but is not specifically applicable to the inoperability of the diverse trip function for the trip breakers. The addition of ITS 3.3.4 Required Action A.2 provides consistent ACTION requirements to compensate for inoperable CRD trip breakers whether or not the inoperability is due to failure of a diverse trip function. The Completion Times in ITS remain, as they are in CTS, significantly different for a CRD trip breaker with an inoperable diverse trip function, as opposed to one which is inoperable for any other reason. ITS Required Action B.1 is added to provide the option of tripping the CRD trip breakers that do not have diverse features inoperable. This additional allowance is less restrictive in that it provides additional flexibility in dealing with an inoperable CRD trip breakers. This is appropriate since tripping the inoperable CRD trip breaker has the same effect as removing power to the CRDMs that are powered through the inoperable CRD trip breakers. The proposed changes are consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change provides Required Actions which allow alternative, equivalent compensatory activities for inoperable equipment to maintain the overall availability of the RPS safety function. This change does not result in any hardware changes. This change

also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the RPS does not change (and therefore any initiation scenarios are not changed), and appropriate response of the RPS continues to be provided by the alternative Required Actions. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required trip devices. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The continued availability of the RPS trip devices is maintained by the proposed change. Therefore, the change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L11

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1 Note j does not include an allowance that allows placing a CONTROL ROD group with an inoperable ETA rely to be placed on a power supply which has OPERABLE ETA relays. ITS 3.3.4 Required Action C.1 is added and provides an alternative to the CTS requirements for inoperable SCR or electronic trip assembly (ETA) relays (CTS Table 3.5.1-1 Note (j)). Required Action C.1 specifically allows for a CONTROL ROD group with an inoperable ETA relay to be placed on a power supply which has OPERABLE ETA relays. This allowance provides new flexibility which is not currently allowed by CTS. Required Action C.1 is an acceptable alternative to opening an inoperable ETA relay because it places the affected CONTROL ROD group on a power supply that ensures the rods are de-energized upon a reactor trip. The change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change provides Required Actions which allow OPERABLE equipment to remain in service to be available to perform its safety function rather than remove it from service due to inoperability of another portion of the channel. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the RPS and CRD trip devices does not change (and therefore any initiation scenarios are not changed), and appropriate response of the RPS and CRD trip devices continues to be provided by the alternative Required Action. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required trip devices. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The continued availability of OPERABLE trip devices is enhanced by the proposed change. Therefore, the change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L12

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1 Column D and Note (e) require the unit be placed in Hot Shutdown within 24 hours and in Cold Shutdown in the following 72 hours when the minimum ES digital actuation logic channels are not OPERABLE. ITS Required Actions Required Action A.1 and A.2 provide two options. One option is to place the associated component(s) in their ES configuration in one hour. The other is to declare them inoperable (and enter into their associated required actions) within one hour. ITS 3.3.7 Required Action A.1 is equivalent to the automatic actuation logic channel performing its safety function ahead of time. Required Action A.2, which requires entry into the Required Action of the affected supported systems, is appropriate since the net result of the automatic actuation logic failure is inoperability of the supported system. The one hour Completion Time reflects the urgency associated with the inoperability of a actuation logic channel which affects multiple safety system components. ITS 3.3.7 is considered less restrictive since declaring the supported systems inoperable associated with one digital channel would result in only one train of the supported system being declared inoperable. At least a 72 hour Completion Time (Refer to ITS 3.5.2 or 3.6.5 Required Actions and Completion Times) is provided to restore one train of ES actuated components to OPERABLE status prior to requiring a unit shutdown. In addition, where practicable, starting the supported system allows continued operation with no further restrictions. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change provides the addition of allowances to place equipment affected by an inoperable ESPS Automatic Actuation Logic Channel in the actuated position or to declare the affected equipment inoperable. This change in ACTION requirements for this instrumentation parameter does not result in any hardware changes.

This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper availability for the required equipment. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The margin of safety for an ESPS digital subsystem is based on availability and capability of the actuated equipment to perform its safety function. This change maintains the capability of the required equipment to perform its safety function even in the absence of its actuating instrumentation. Therefore, this change does not represent a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L13

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.5.1.1 requires the source range and wide range instruments to be OPERABLE in a startup mode or in a critical state. CTS Table 3.5.1-1 Note b indirectly provides a qualification to this statement of Applicability. This note provides a relaxation of action requirements when "2 of 4 power range instrument channels are greater than 10% rated power." The Applicability of ITS 3.3.9 and 3.3.10 does not require either the source range instrument channel or the wide range instrument channel to be maintained OPERABLE above MODE 2. This represents a relaxation of requirements, by removing the requirement to take actions in the event that either the required source range instrument channel or the required wide range instrument channel is inoperable, when above 5% RTP (ITS) but less than or equal to 10% rated power, as indicated on the power range instruments (CTS). This is acceptable since the power range channels provide all assumed reactor protection above 5% RTP. This change is being made to provide clear statements of Applicability for these specifications which are consistent with the requirements of the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The Applicability for the source range and wide range instrument channels of RPS is limited such that they are not required above MODE 2. Similarly, the Required Actions have been revised such that no actions are required if these channels are inoperable in MODE 1. This change in Applicability and Required Actions for these functions does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis.

Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure appropriate availability for the instrument channels considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The instrument channels provide neutron power indication and control rod withdrawal inhibit interlocks (based on high startup rate), under low power operating conditions. In the conditions to be excluded from the Applicability, indication of neutron power is provided by the power range instrumentation channels. Therefore, the changes do not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L14

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS Table 4.1-1 requires a calorimetric heat balance check and adjustment every shift. ITS SR 3.3.1.2, which requires the verification every 24 hours, is modified by a Note that allows this check be delayed as much as 24 hours after THERMAL POWER is ≥ 15% RTP. The ITS recognizes the difficulty in performing the heat balance check and the limitations of the calorimetric while operating at very low power levels. No specific allowance is provided in the CTS which removes the requirement to perform this calibration when in a critical state at low power levels. Below 15% RTP, ONS calculates heat balance power level based totally upon the primary system parameters. Above 15% RTP, the secondary system parameters are also considered since they are generally more accurate at higher power levels. By allowing the delay in performance of this calibration until RTP is above 15%, a generally more accurate calorimetric (one including secondary system parameters) is available. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   A Note is included which allows deferring the calorimetric heat balance for adjustment of the power range instrument channels of the RPS while at low power levels. This change in applicability for this Surveillance does not result in any hardware changes. The power range monitors are not considered as initiators for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. Performance of this Surveillance at low power levels generally provides less accurate results than at higher power levels. Since the results of the Surveillance are typically small adjustments, the change which allows nonperformance during low power does not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure appropriate availability for the instrument channels considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The power range instrument channels provide no identifiable margin of safety at low power since their calibration to the calorimetric heat balance does not provide accurate results. In the conditions to be excluded from the Surveillance, the power range instrumentation is available, but calibration is recognized as uncertain. Therefore, the change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L15

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS Table 4.1-1 requires a comparison of the out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE every 31 days. ITS SR 3.3.1.3, which provides an equivalent requirement, is modified by a Note that allows a delay in performance of this SR until the unit is above 15% RTP. This allowance is appropriate due to the usable range of the incore nuclear instruments which are required for the performance of this SR. Below about 15% the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. Adoption of this Note provides a specific relaxation of requirements where none existed in CTS. This change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   A Note is included which allows deferring the comparison and adjustment of the power range instrument channels of the RPS against the incore detectors while at low power levels. This change in applicability for this Surveillance does not result in any hardware changes. The power range monitors are not considered as initiators for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. Performance of this Surveillance at low power levels generally provides less accurate results than at higher power levels. Since the results of the Surveillance are typically small adjustments, the change which allows nonperformance during low power does not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure appropriate availability for the instrument channels considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   In the conditions to be excluded from the Surveillance, the power range instrumentation is available, but calibration is recognized as imprecise. Since the proposed change does not affect the OPERABILITY of the power range instrumentation and the Surveillance is only deferred to a power level in which an accurate calibration can be performed, the change does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L16

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.4.2, which requires the automatic initiation circuitry associated with loss of main feedwater pumps to be OPERABLE prior to criticality, provides no allowed outage time when one of two loss of main feedwater instrumentation channels are inoperable. ITS 3.3.14, ACTION A is added to allow continued reactor operation for an indefinite period when one of two EFW System loss of main feedwater instrumentation channels in an EFW pump automatic initiation circuit is inoperable provided the inoperable channel is placed in a tripped condition for initiation within one hour. ITS 3.3.14 ACTIONS Note is added to allow separate condition entry for each EFW pump initiation circuit. This allows one hour to place the channel in trip for each function when Condition A is entered. This less restrictive provision is acceptable since this leaves the function in a one-out-of-one logic configuration for initiation versus the normal two-out-of-two logic configuration. This maintains at least equivalent reliability for EFW initiation. EFW is maintained single failure proof by the separate initiation circuits for each the three EFW pumps. This less restrictive change is consistent with NUREG Specification 3.3.11, ACTION A.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change allows indefinite continued operation with one required channel inoperable provided it is placed in a tripped condition within one hour. This action leaves the system in a one-out-of-one condition for actuation. Thus, if another channel were to fail, the EFW instrumentation can still perform its initiation functions. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the EFW instrumentation does not change (and therefore any initiation scenarios are not changed). Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that originally

considered. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The change ensures proper availability for the required EFW initiation functions. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   This change to the EFW instrumentation requirements does not involve a change in setpoints and cannot affect any margin of safety associated with the response to a design basis accident. The change does not prevent the EFW instrumentation from performing their actuation functions since the action places the ESPS instrumentation in a one-out-of-one condition for initiation versus the normal two-out-of-two logic. Thus, if another channel were to fail, the EFW instrumentation could still perform its initiation functions. Therefore, this change to allow the EFW initiation functions to operate indefinitely with one required EFW instrument channel inoperable provided the channel is placed in the tripped condition within one hour, is not considered to involve a significant reduction in the margin of safety

LESS RESTRICTIVE CHANGE L17

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

CTS 3.7.6 and 3.7.7 both require an inoperable voltage sensing relay to be restored within 72 hours (Required Action A.1). ITS 3.3.19 Required Action A.1 and 3.3.20 Required Action A.1 require the inoperable channel to be placed in trip within 72 hours. This less restrictive change allows operation to continue indefinitely when the channel is placed in trip and continues to allow 72 hours to restore an inoperable channel that cannot be placed in trip. The actuation logic for DGVP is two-out-of-three. Placing the inoperable channel in the tripped condition fulfills the function of the channel (and places the function in a one-out-of-two configuration). Indefinite operation in this configuration is acceptable since the degraded grid voltage function is capable of performing its function in the presence of a single failure. This change is consistent with comparable NUREG 3.3.8 requirements.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change allows indefinite continued operation with one voltage sensing channel inoperable, provided the inoperable voltage sensing channel is placed in trip within 72 hours. This action leaves the system in a one-out-of-two condition for actuation. Thus, if another channel were to fail, the DGVP instrumentation can still perform its function. This change does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the DGVP instrumentation does not change (and therefore any initiation scenarios are not changed). Also, the change does not change the assumed response of the equipment in performing its specified function from that originally considered. Therefore, the changes do not significantly increase the consequences of an accident.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The change ensures proper availability for the required DGVP function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   This change to the DGVP instrumentation requirements does not involve a change in setpoints and cannot affect any margin of safety associated with the response to a design basis accident. The change does not prevent the DGVP instrumentation from performing their function since the action places the DGVP instrumentation in a one-out-of-two condition for actuation versus the normal two-out-of-three logic. Thus, if another channel were to fail, the DGVP instrumentation could still perform its initiation functions. Therefore, this change to allow the DGVP initiation functions to operate indefinitely with one required DGVP instrument channel inoperable provided the channel is placed in the tripped condition within 72 hours, is not considered to involve a significant reduction in the margin of safety

LESS RESTRICTIVE CHANGE L18

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.5.7 Applicability for the Main Steam Line Break and Feedwater Isolation Circuity is when main steam header pressure is greater than 700 psig. ITS 3.3.11, 12, & 13 Applicability is MODES 1 and 2, and MODE 3 with main steam header pressure greater than 700 psig except when all MFCVs and SFCVs are closed. The exception of "when all MFCVs and SFCVs are closed" is a less restrictive change and is consistent with comparable NUREG requirements (Table 3.3.11-1, Note d). The exception is appropriate since the MFCVs and SFCVs are already performing their safety function when they are closed.
>
> Required Action B.2.2 of ITS 3.3.11, 12 and 13 is added to provide the option of closing the MFCVs and SFCVs in lieu of reducing main steam header pressure to less than 700 psig. This optional allowance is consistent with the applicability since closure of the MFCVs and SFCVs removes the unit from the Applicability of the LCO.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The MSLB and MFW Isolation circuitry is not an initiator of analyzed events. Therefore, the probability of an accident is independent of the status of the MSLB and MFW Isolation circuitry. As such the proposed change does not involve a significant increase in the probability of an accident previously evaluated. The proposed change eliminates the requirement for MSLB and MFW Isolation circuitry OPERABILITY when all the MFCVs and SFCVs are closed. When the MFCVs and SFCVs are closed the MSLB and MFW Isolation circuitry has no safety function since its function is to close the MFCVs and SFCVs when conditions indicate a MSLB. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Since MSLB and MFW Isolation circuitry requirements continue to require OPERABILITY when the reactor is in a condition that requires their function, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L19

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1 requires a unit shutdown within 24 hours when one or more turbine stop valve closure channels is inoperable. ITS 3.3.15 ACTION A is added to allow one hour to declare the associated TSVs inoperable. The additional hour allowed to restore the instrumentation channel(s) prior to requiring further action is consistent with similar NUREG Required Actions that require supported equipment to be declared inoperable (e.g., NUREG Specification 3.3.7, Required Action A.2). The one hour Completion Time is considered sufficient to correct minor problems. Even with the 1 hour, the unit gets to subcriticality sooner (13 hours) than that time allowed by CTS Table 3.5.1-1, Column D for Item 16 (24 hours).

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change establishes a 1 hour Completion Time during which the unit may continue operation with MSLB and MFW Isolation instrumentation inoperable. This change provides an opportunity to repair the inoperable instrumentation channel(s) prior to declaring the equipment supported by it inoperable. The addition of this allowed condition with a short Completion Time does not result in any hardware changes. The allowed condition also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Further, the consequences of an accident are the same during the additional one hour time period allowed for instrument channel restoration as it is during the time period currently allowed for restoring TSVs to OPERABLE status. Therefore, the change does not significantly increase the probability of occurrence of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The change continues to ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The allowed condition has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, this new allowed condition does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L20

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-1 calibration requirements for RPS functions that receive input from neutron detectors do not specifically exclude the detectors from the calibration of that function. ITS SR 3.3.1.6, which provides comparable CHANNEL CALIBRATION requirements for RPS functions, includes a note that specifically excludes the neutron detectors from the CHANNEL CALIBRATION. This exclusion is appropriate because of the passive design of the detectors, the extreme difficulty in both accessing the detectors and in generating an appropriate input signal to the detectors, the fact that no specific adjustments can be made to the detectors, and the principles of detector operation that ensure a virtually instantaneous response. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change excludes the neutron detectors from the CHANNEL CALIBRATION requirements for RPS functions that receive input from the detectors. The probability of an accident is not increased by these changes because the proposed change does not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, or modified. The consequences of an accident will not be increased because the change will not affect the ability of the power range detectors to monitor and respond to core conditions. Changes in neutron detector sensitivity are compensated for by performance of the 24 hour heat balance check and adjustment of SR 3.3.1.2. The neutron detectors are excluded from the CHANNEL CALIBRATIONS because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed). The equipment function has not changed, nor has its interface with other equipment. Adequate assurance continues to be provided to ensure the neutron detectors remain capable of performing their function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The proposed change excludes neutron detectors from CHANNEL CALIBRATION Surveillance Requirements. The proposed change does not involve a significant reduction in a margin of safety because the change will not affect the ability of the Power Range detector to monitor and respond to core conditions. The neutron detectors are excluded from the CHANNEL CALIBRATION because they are passive devices, with minimal drift, and because of the difficulty of simulating a meaningful signal. Changes in neutron detector sensitivity are compensated for by performance of the 24 hour heat balance check and adjustment of SR 3.3.1.2. As a result, the change does not affect the current analysis assumptions and adequate assurance is provided that the neutron detectors will be maintained OPERABLE. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L21

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.6-1 Action 3 for the Reactor Vessel Head Level and the Reactor Vessel Level (ITS 3.3.8 PAM #3 and #5) allows, if repairs are feasible, 7 days for restoration of a single inoperable instrument channel when one or both instrument channels are inoperable. Operation may continue with one inoperable channel, provided a report is submitted within the next 30 days outlining the cause of the inoperability and the plans and schedule for restoring the channel to OPERABLE status. When both are inoperable, if at least one instrument channel is not restored, the unit is then required to be in hot shutdown within 12 hours. ITS ACTION A allows 30 days for restoration of a single channel, and ITS ACTION C allows 7 days for restoration of one of two inoperable instrument channels. ITS ACTIONs B and I then require a Special Report. Therefore, the proposed Required Action I.1 is less restrictive since a unit shutdown is not required. Required Action I.1 is appropriate in lieu of a shutdown requirement since both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling capability and both the subcooling monitoring monitors and core exit thermocouples provide an alternate means of monitoring for this purpose. This change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change does not result in any hardware changes, but does allow additional continued operation with inoperable Reactor Vessel Head Level instruments. These instruments provide indication only and are not considered as initiators of any analyzed event. Therefore, the change does not significantly increase the probability of occurrence of any previously analyzed event. Neither will the change result in a significant increase in the consequences of any accident previously evaluated since the consequences of an event occurring during the operation of the unit during the Completion Times are the same as the consequences

of an event occurring while operating under the current ACTIONS. Therefore, the change does not involve a significant increase to the consequences of any accident previously evaluated

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed). The equipment function has not changed, nor has its interface with other equipment. The change still ensures proper actions are required, consistent with applicable regulatory guidance. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The margin of safety for PAMs is based on availability and capability of the instrumentation to provide the required information to the operator. The availability and capability of the PAMs may be affected but is not considered to be significant due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L22

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> The CTS Actions (Table 3.5.6-1, Action 1) for the Containment Pressure -High Range (PAM #7) Function allow 7 days for restoration of an inoperable instrument channel and 48 hours for restoration of an inoperable channel when both are inoperable. When either required action is not met, the unit must be placed in hot shutdown within the next 12 hours. ITS 3.3.8, ACTION A allows 30 days for restoration of a single channel. If not restored, then a Special Report is required by Required Action B.1. This is less restrictive since a unit shutdown is not required. ITS Required Action B.1 is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. ITS ACTION C allows an additional 5 days for restoration of a single channel when both channels are inoperable. The additional time allowed to restore at least one channel allowed by Action C is considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. These less restrictive changes are consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change does not result in any hardware changes, but does allow additional continued operation with inoperable Containment Pressure - High Range instruments. These instruments provide indication only and are not considered as initiators of any analyzed event. Therefore, the change does not significantly increase the probability of occurrence of any previously analyzed event. Neither will the change result in a significant increase in the consequences of any accident previously evaluated since the consequences of an event occurring during the operation of the unit during the Completion Times are the same as the consequences

of an event occurring while operating under the current ACTIONS. Therefore, the change does not involve a significant increase to the consequences of any accident previously evaluated

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed). The equipment function has not changed, nor has its interface with other equipment. The change still ensures proper actions are required, consistent with applicable regulatory guidance. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for PAMs is based on availability and capability of the instrumentation to provide the required information to the operator. The availability and capability of the PAMs may be affected but is not considered to be significant due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L23

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

The CTS Actions (Table 3.5.1-1, Action 2) for Containment Water Level (PAM #6), Containment High-Range Radiation (PAM #9), Containment Hydrogen (PAM #10), and the Core Exit Thermocouple (PAM #16) Functions allow 30 days for restoration of an inoperable instrument channel and 48 hours for restoration of an inoperable channel when both are inoperable. When either required action is not met, the unit must be placed in hot shutdown within the next 12 hours. ITS 3.3.8, ACTION A allows 30 days for restoration of a single channel and then a Special Report is required by Required Action B.1. ITS Required Action B.1 is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. This is less restrictive since a unit shutdown is not required. ITS ACTION C for the Containment Water Level, Containment High-Range Radiation, and the Core Exit Thermocouple Functions allows an additional 5 days for restoration of a single channel when both channels are inoperable. ITS ACTION D for the Containment Hydrogen Concentration Function allows an additional 24 hours. The additional time allowed to restore at least one channel allowed by ITS Actions C and D are considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. ITS 3.3.8, Action I is added for ITS Table 3.3.8-1, Function 9 (Containment High-Range Radiation), to allow a Special Report in place of the CTS requirement for shutdown. This is acceptable since alternate means are available to monitor this variable. These changes are consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change does not result in any hardware changes, but does allow additional continued operation with inoperable PAM functions. These instruments provide indication only and are not

considered as initiators of any analyzed event. Therefore, the change does not significantly increase the probability of occurrence of any previously analyzed event. Neither will the change result in a significant increase in the consequences of any accident previously evaluated since the consequences of an event occurring during the operation of the unit during the Completion Times are the same as the consequences of an event occurring while operating under the current ACTIONS. Therefore, the change does not involve a significant increase to the consequences of any accident previously evaluated

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed). The equipment function has not changed, nor has its interface with other equipment. The change still ensures proper actions are required, consistent with applicable regulatory guidance. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for PAMs is based on availability and capability of the instrumentation to provide the required information to the operator. The availability and capability of the PAMs may be affected but is not considered to be significant due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L24

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> The CTS Actions (Table 3.5.1-1, Action 4) for the Subcooling Monitor function (PAM #17) Function allows 30 days for restoration of an inoperable instrument channel and 48 hours for restoration of an inoperable channel when both are inoperable. When either required action is not met, the unit must be placed in hot shutdown within the next 12 hours. ITS 3.3.8, ACTION A allows 30 days for restoration of a single channel and then a Special Report is required by ITS Required Action B.1. ITS Required Action B.1 is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. This is less restrictive since a unit shutdown is not required. ITS ACTION C allows an additional 5 days for restoration of a single channel when both channels are inoperable. The additional time allowed to restore at least one channel allowed by ITS Action C is considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. These changes are consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change does not result in any hardware changes, but does allow additional continued operation with inoperable Subcooling Monitor instrument channels. This PAM function provides indication only and is not considered as an initiator of any analyzed event. Therefore, the change does not significantly increase the probability of occurrence of any previously analyzed event. Neither will the change result in a significant increase in the consequences of any accident previously evaluated since the consequences of an event occurring during the operation of the unit during the Completion Times are the same as the consequences of an event occurring while operating under the current ACTIONS.

Therefore, the change does not involve a significant increase to the consequences of any accident previously evaluated

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed). The equipment function has not changed, nor has its interface with other equipment. The change still ensures proper actions are required, consistent with applicable regulatory requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The margin of safety for PAMs is based on availability and capability of the instrumentation to provide the required information to the operator. The availability and capability of the PAMs may be affected but is not considered to be significant due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L25

Not used.

LESS RESTRICTIVE CHANGE L26

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-1 requires a CHANNEL CHECK of items 5, 26, 30, and 39 either shiftly or weekly. ITS SR 3.3.8.1 requires a CHANNEL CHECK of PAM instrument channels for each required channel that is normally energized every 31 days. The Frequency is based on operating experience that demonstrates channel failure is rare, and on the use of less formal but more frequent checks of channels during normal operational use of the displays associated with the required channels. This less restrictive change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The PAM instruments are used to support mitigation of the consequences of an accident; however, they are not considered the initiator of any previously analyzed accident. As such, the revision of the Surveillance Frequency of the PAMs does not increase the probability of any accident previously evaluated. Since the function of the PAM instruments continues to be verified, and continues to be required to be OPERABLE, the change of the Surveillance Frequency will not reduce the capability of required equipment to mitigate the event. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The change still ensures proper surveillances are required for the equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for PAMs is based on availability and capability of the instrumentation to provide the required information to the operator. The Frequency is based on operating experience that demonstrates channel failure is rare, and on the use of less formal but more frequent checks of channels during normal operational use of the displays associated with the required channels. Therefore, the availability and capability of the PAMs continues to be assured by the Surveillance Frequency and this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L27

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-1, items 54 and 57, requires a monthly functional test of the containment high range radiation monitor and containment hydrogen monitor instrument channels. This monthly functional test is not included in ITS. Such a test is typically required when the instrumentation provides a safety related automatic actuation function. This instrument channel provides information only, and as such, a CHANNEL FUNCTIONAL TEST is not appropriate, nor required. This change is also consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The PAM instruments are used to support mitigation of the consequences of an accident; however, they are not considered the initiator of any previously analyzed accident, nor do they provide any automatic actuation functions. As such, the revision to omit the Surveillance Requirement for functional testing of the PAM instruments does not increase the probability of any accident previously evaluated. Since the capability of the PAM instruments to provide the required information continues to be verified, and continues to be required to be OPERABLE, the change will not reduce the capability of required equipment to mitigate the event. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The change still ensures proper surveillances are required for the equipment considered in the safety analysis. Thus, this change

does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for PAMs is based on availability and capability of the instrumentation to provide the required information to the operator. The Frequency is based on operating experience that demonstrates channel failure is rare, and on the use of less formal but potentially more frequent checks of channels during normal operational use of the displays associated with the required channels. Therefore, the availability and capability of the PAMs continues to be assured by the proposed Surveillance Requirements and this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L28

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1, Column D requires the unit be placed in hot shutdown within 12 hours when less than two source range channels are OPERABLE and rated power is ≤ 10% as shown on the power range channels and ≤ 4 x 10^{-4} % rated power as shown on the wide range channels. Comparable ITS Required Actions do not require the unit be placed in hot shutdown. Therefore, the proposed ITS Required Actions are less restrictive in this aspect. However, other more appropriate required actions are added to replace the CTS required action (Refer to DOC M13). This change is also consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The source range instrument channels are the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. The source range instrument channels are not initiators of analyzed events. Therefore, the probability of an accident is independent of the status of the source range instrumentation. The proposed change eliminates the CTS requirement to place the unit in hot shutdown when the minimum required channels are not OPERABLE. This is acceptable since other more appropriate required actions are added to limit the possibilities for adding positive reactivity. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant

operation. The proposed change will still ensure proper availability for the required source range instrument functions. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

Since the required actions for inoperable source range instruments continued to limit the possibilities for adding positive reactivity and require a check of shutdown margin periodically, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L29

Not used.

LESS RESTRICTIVE CHANGE L30

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.5.1.5 requires the overlap between the wide range and the source range instrumentation to be checked during startup. Proposed ITS SR 3.3.10.3 requires the overlap to be verified every startup if not performed within the previous 7 days. The ITS allows the test to be omitted if performed within the previous 7 days. This is based on industry operating experience which shows the instrument overlap does not change appreciably within this test interval. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The wide range instrument channels are not assumed to be initiators of any analyzed event. This change eliminates the requirement to verify the overlap between the wide range and the source range instrumentation during startup when the verification has been performed within the previous 7 days based on operating experience which shows the overlap does not change appreciably within this test interval. The change removes an unnecessary additional performance of a surveillance which has been performed within its normal Frequency. Not performing the surveillance prior to startup would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore the probability and consequence of an accident previously evaluated are not significantly increased.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change continues to ensure overlap between the source range and wide range instrumentation channels is verified. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Based on operating experience, changes in source range and wide range instrument overlap does not change appreciably within the 7 day test interval. Consequently, the proposed Frequency is sufficient to identify significant changes. Therefore, eliminating the requirement to verify overlap during every startup does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L31

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1, Column D requires the unit be placed in hot shutdown within 12 hours when less than two wide range channels are OPERABLE. ITS 3.3.10, Required Action A.1 only requires that power be reduced to $< 4 \times 10^{-4}$% RTP when one channel is inoperable. The proposed change is less restrictive since the CTS defines Hot Shutdown as the reactor having a K_{eff} of ≤ 0.99 and the reactor could have a K_{eff} of > 0.99 with power reduced below 4×10^{-4}% RTP as allowed by Required Action A.1. The proposed change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The wide range instrumentation is designed to detect power changes during initial criticality and power escalation when the power range channels cannot provide reliable indication. The wide range instrument channels are not initiators of analyzed events. Therefore, the probability of an accident is independent of the status of the wide range instrumentation. As such the proposed change does not involve a significant increase in the probability of an accident previously evaluated. The proposed change requires reactor power to be reduced to $< 4 \times 10^{-4}$ when the minimum required channels are not OPERABLE in lieu of the CTS requirement to place the unit in Hot Shutdown (K_{eff} of ≤ 0.99). At power levels below 4×10^{-4}, the source range instruments become the primary means for monitoring reactivity. Since the required action results in the unit being placed in a condition in which reliable indication continues to be provided, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Since the required actions for an inoperable wide range instrument channel requires a reduction in power to the point where source range channels can provide neutron flux indication, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L32

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 4.1-1, Column "Test," Items 5 and 6 require a functional test be performed on the source range and wide range channels prior to startup. This requirement is not retained in the ITS. Consistent with the NUREG, a CHANNEL CALIBRATION of the source range and wide range instruments is added (Refer to DOC M14). Because the calibration by definition encompasses the functional test, performance of the calibrations will ensure that testing is consistent with CTS requirements. The frequency of this testing is now based strictly on the time since its last performance and not dependent upon whether or not the unit is in startup. This change is acceptable, based on operating experience which demonstrates the source and wide range instruments are highly reliable.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The change replaces a functional test which is required to be performed "prior to startup" with a CHANNEL CALIBRATION on an 18 month Frequency. This change does not result in any hardware changes or changes in operating methods. The source range and wide range instrumentation are not considered as the initiator of any previously analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of any accident previously evaluated. Additionally, neither the test, nor the test Frequency impact the operation of equipment or its response to any event. Therefore, the proposed change does not involve a significant increase in the consequences of any accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change continues to ensure adequate surveillance is performed to identify any degradation of the source range and wide range instrumentation channels. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The margin of safety for source range and wide range instrument channels is based on availability and capability of the instrument to perform its safety function. If the unit operates with only one "startup" per cycle, the Frequency for these surveillances is the same, but the calibration would be an additional requirement because it includes testing activities in addition to the functional test. Industry performance history of this type of instrumentation has demonstrated reliability of the equipment over an operating cycle. Therefore, a periodic Frequency of 18 months has been determined to be adequate to confirm the availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L33

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.7.5.1 requires performance of SR 3.7.1.11 (Keowee emergency start) and SR 3.7.1.14 (EPSL automatic transfer). SR 3.7.1.11 verifies that each Keowee Hydro Unit (KHU) can emergency start from each control room, attain rated speed and voltage within 23 seconds of an emergency start initiate, and be synchronized to the grid and loaded. The test is performed by manually starting one KHU from the Unit 1 and 2 Control Room and the other KHU from the Unit 3 Control Room. The accident analyses do not take credit for a manual Keowee start during operation above Cold Shutdown. Therefore, the requirement to test this function during operation above Cold Shutdown is not retained. This function is required to be OPERABLE during MODES 5 and 6 and during movement of irradiated fuel assemblies by ITS 3.3.22, "EPSL Manual Keowee Emergency Start Function."

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The requirement to test the manual Keowee Emergency Start Function during operation above Cold Shutdown is not retained. Since this function is not an initiator of any analyzed event, the probability of an accident is not significantly increased. The accident analyses do not take credit for a manual Keowee start during operation above Cold Shutdown, therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a

new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Since the accident analyses do not take credit for a manual Keowee start during operation above Cold Shutdown, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L34

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.5.1.1 Applicability for the TSV Closure instrumentation channels is while in the startup mode or when the reactor is in a critical state. This is considered encompassed by ITS MODES 1 and 2. ITS 3.3.15 Applicability is in MODES 1, 2, and 3 except when all TSVs are closed. The exception of "when all TSVs are closed" is a less restrictive change and is consistent with comparable NUREG requirements (Table 3.3.11-1, Note c). The exception is appropriate since the TSVs are already performing their safety function when they are closed.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The TSV Closure instrument channels are not initiators of analyzed events. Therefore, the probability of an accident is independent of the status of the TSV Closure instrumentation. As such the proposed change does not involve a significant increase in the probability of an accident previously evaluated. The proposed change eliminates the requirement for TSV closure instrument OPERABILITY when all the TSVs are closed. When the TSVs are closed the TSV Closure instrumentation has no safety function since its function is to close the TSVs on a reactor trip. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   Since TSVs closure instrumentation requirements continues to require OPERABILITY when the reactor is in a condition that requires their function, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGE L35

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.8.10 requires the radiation monitor associated with the purge system valve isolation to be tested and verified OPERABLE immediately prior to refueling operations. CTS Table 4.1-2, Item 4, requires this functional test be performed "Prior to Refueling." ITS 3.3.16 Applicability is during CORE ALTERATIONS and during movement of irradiated fuel assemblies within containment. ITS SR 3.3.16.2 requires the testing be performed once each refueling outage prior to CORE ALTERATIONS or beginning movement of irradiated fuel assemblies within containment. Permitting the specified testing to be conducted prior to beginning movement of irradiated fuel assemblies within containment in lieu of immediately prior to refueling operations is a less restrictive requirement upon unit operation (and is more stringent than the NUREG). Requiring performance of SR 3.3.16.2 once each refueling outage prior to CORE ALTERATIONS or prior to beginning movement of irradiated fuel assemblies within containment represents a reasonable relaxation of the CTS surveillance frequency. This continues to ensure that this function is verified prior to irradiated fuel assembly handling within containment.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The isolation function of the radiation monitor associated with the purge system valves is not assumed to be an initiator of any analyzed event. As a result, the probability of an accident occurring is independent of the status of testing the isolation function of the radiation monitor associated with the purge system valves. This change eliminates the requirement for testing of this isolation function immediately prior to refueling operations. The change continues to require the isolation function to be

OPERABLE and continues to ensure that this function is verified within a reasonable interval prior to irradiated fuel assembly handling within containment. This provides reasonable assurance the isolation function of the radiation monitor associated with the purge system valves remains OPERABLE. Therefore the consequence of an accident previously evaluated are not significantly increased.

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still require the isolation function of the radiation monitor associated with the purge system valves be OPERABLE. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The isolation function of the radiation monitor associated with the purge system valves is still required to be OPERABLE. This change continues to ensure that this function is verified within a reasonable interval prior to irradiated fuel assembly handling within containment. Therefore the margin of safety has not been significantly reduced.

LESS RESTRICTIVE CHANGE L36

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS Table 3.5.1-1, Column D requires the unit to be in hot shutdown within 24 hours when one or more TSV Closure Instrumentation channels is inoperable and Note (e) to the Table requires the unit be placed in Cold Shutdown within the following 72 hours if the minimum conditions are not met. ITS 3.3.15 ACTION A is added to require the TSVs to be declared inoperable within 1 hour (also, see DOC L19). ITS 3.7.2, Turbine Stop Valves, then dictates the required action for inoperable TSVs. With one or more TSVs inoperable in MODE 1, Required Action A.1 requires the TSVs be restored to OPERABLE status within 8 hours or Required Action B.1 requires the unit be in MODE 2 in 6 hours. Therefore, this portion of ITS is more restrictive since the unit must be in MODE 2 within 15 hours of an inoperable TSV Closure instrumentation channel where CTS required the unit be in hot shutdown (equivalent to ITS MODE 3) within 24 hours. ITS 3.7.2 Action C allows 8 additional hours to close an inoperable TSV when in MODE 2 or 3 (total of 23 hours). In addition, if it were not closed, then an additional 12 hours (on top of the eight hours) is allowed to place the unit in MODE 3 and 18 hours to place the unit in MODE 4. This results in allowing a total of 35 hours to be in MODE 3 and 41 hours to be in MODE 4 from initial discovery of it being inoperable in MODE 1. This compares to the CTS time allowed to place the unit in Hot Shutdown (MODE 3) of 24 hours. Therefore, an additional 11 hours is allowed to place the unit in MODE 3. The additional time is reasonable considering the low probability of an accident occurring during this time period that would require closure of the TSVs. The more restrictive aspects of this change are addressed in DOC M22. The proposed less restrictive ITS Shutdown Times requirements are consistent with ITS 3.7.2, which is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change does not result in any hardware changes, but does allow additional time to close the TSVs and for unit shutdown with inoperable TSV Closure instrument channels. The TSV Closure function is not an initiator of any analyzed event. Therefore, the change does not significantly increase the probability of occurrence of any previously analyzed event. Neither will the change result in a significant increase in the consequences of any accident previously evaluated since the consequences of an event occurring during the time period allowed for Shutdown is the same as the consequences of an event occurring during the current time period allowed for shutdown. Therefore, the change does not involve a significant increase to the consequences of any accident previously evaluated

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed). The equipment function has not changed, nor has its interface with other equipment. The change still ensures proper actions are required, consistent with applicable regulatory requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

   The additional time allowed to reach hot shutdown (equivalent to ITS MODE 3) is acceptable based on the small probability of an event requiring the inoperable Technical Specification component to function. As such, the extension from 24 hours to 35 hours for MODE 3 does not involve a significant reduction in the margin of safety.

LESS RESTRICTIVE CHANGE L37

The Oconee Nuclear Station is converting to the Improved Technical Specifications (ITS) as outlined in NUREG-1430, "Standard Technical Specifications, Babcock and Wilcox Plants." The proposed changes involve making the current Technical Specifications (CTS) less restrictive. Below is the description of this less restrictive change and the No Significant Hazards Consideration for conversion to NUREG-1430.

> CTS 3.4.3.b requires a flow path with no OPERABLE emergency feedwater flow indicators to be restored to OPERABLE status within 72 hours. ITS 3.3.8 Required Action C.1 allows 7 days to restore an inoperable flow indicator when both are inoperable. Required Action C.1 allows an additional 4 days for restoration of a single channel when no channels are OPERABLE. The additional time to restore at least one channel allowed by Required Action C.1 is considered appropriate based on the relatively low probability of an event requiring PAM instrumentation and the availability of alternate means to obtain required information. This less restrictive change is consistent with the NUREG.

In accordance with the criteria set forth in 10 CFR 50.92, Duke Energy has evaluated this proposed Technical Specification change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. **Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   This change does not result in any hardware changes, but does allow additional continued operation with inoperable EFW flow instruments. These instruments provide indication only and are not considered as initiators of any analyzed event. Therefore, the change does not significantly increase the probability of occurrence of any previously analyzed event. Neither will the change result in a significant increase in the consequences of any accident previously evaluated since the consequences of an event occurring during the operation of the unit during the Completion Times are the same as the consequences of an event occurring while operating under the current ACTIONS. Therefore, the change does not involve a significant increase to the consequences of any accident previously evaluated

2. **Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?**

   The change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed). The equipment function has not changed, nor has its interface with

other equipment. The change still ensures proper actions are required, consistent with applicable regulatory guidance. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. **Does this change involve a significant reduction in a margin of safety?**

The margin of safety for PAMs is based on availability and capability of the instrumentation to provide the required information to the operator. The availability and capability of the PAMs may be affected but is not considered to be significant due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments. Therefore, this change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT

This proposed Technical Specification Change has been evaluated against the criteria for and identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.21. It has been determined that the proposed changes meet the criteria for categorical exclusion as provided for under 10 CFR 51.22 (c) (9). The following is a discussion of how the proposed Technical Specification Change meets the criteria for categorical exclusion.

10 CFR 51.22 (c) (9): Although the proposed change involves changes to requirements with respect to inspection or surveillance requirements;

(i) the proposed change involves no Significant Hazards Consideration (refer to the No Significant Hazards Consideration section of this Technical Specification Change Request),

(ii) there is no significant change in the types or significant increase in the amounts of any effluents that may be released offsite since the proposed changes do not affect the generation of any radioactive effluents nor do they affect any of the permitted release paths, and

(iii) there is no significant increase in individual or cumulative occupational radiation exposure.

Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22 (c)(9). Based on the aforementioned and pursuant to 10 CFR 51.22 (b), no environmental assessment or environmental impact statement need be prepared in connection with issuance of an amendment to the Technical Specifications incorporating the proposed changes of this request.

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.3 - INSTRUMENTATION

ATTACHMENT 5

NUREG 1430 MARKUP AND JUSTIFICATIONS

TECHNICAL SPECIFICATIONS

3.3 INSTRUMENTATION

4

3.3.1 Reactor Protection System (RPS) Instrumentation

ve

Three 6

LCO 3.3.1 ~~Four~~ Three channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE. 3.5.1.1 T3.5.1-1, Col. C

APPLICABILITY: According to Table 3.3.1-1. 3.5.1.1 DOC M2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
~~A. One channel inoperable.~~	~~A.1 Place channel in bypass or trip.~~	~~1 hour~~	6
A ~~B~~. ~~Two~~ One required channel(s) inoperable. 6	A ~~B~~.1 Place ~~one~~ channel in trip. 6	1 hour	T3.5.1-1, Note (a)
26 Two or more required channels inoperable	~~AND A B.2 Place second channel in bypass.~~	~~1 hour~~	6
OR B ~~C~~. Required Action and associated Completion Time of Condition A ~~or B~~ not met. 6	B ~~C~~.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately	3.5.1.2
C ~~D~~. As required by Required Action B ~~C~~.1 and referenced ~~in~~ Table 3.3.1-1. 6	C ~~D~~.1 Be in MODE 3. AND C ~~D~~.2 Open all ~~CONTROL ROD~~ drive (CRD) trip breakers. 2	12 ~~6~~ hours 5 12 ~~6~~ hours	T3.5.1-1, Col. D DOC M1

(continued)

ACTIONS (continued)

6 CONDITION	6 REQUIRED ACTION	COMPLETION TIME	
D ~~E~~. As required by Required Action B ~~C~~.1 and referenced in Table 3.3.1-1.	D ~~E~~.1 Open all CRD trip breakers.	6 hours	DOC M30
E ~~F~~. As required by Required Action B ~~C~~.1 and referenced in Table 3.3.1-1.	E ~~F~~.1 Reduce THERMAL POWER < ~~[45]~~ 30 % RTP. 46	6 hours	T3.5.1-1, COL. D DOC L2
F ~~G~~. As required by Required Action B ~~C~~.1 and referenced in Table 3.3.1-1.	F ~~G~~.1 Reduce THERMAL POWER < ~~[15]~~ 2 % RTP. 46	12 ~~6~~ hours 5	T3.5.1-1, Col. D DOC L2

SURVEILLANCE REQUIREMENTS

------------------------------------NOTE------------------------------------
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function. 4.1.1

SURVEILLANCE	FREQUENCY	
SR 3.3.1.1 Perform CHANNEL CHECK.	12 hours	T 4.1-1 Col. "Check"

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.3.1.2	NOTE: Not required to be performed until ~~[24]~~ hours after THERMAL POWER is ≥ 15% RTP. (1) Compare results of ~~Verify~~ calorimetric heat balance ~~is~~ to the calculation (1) ~~≤ [2]% RTP greater than~~ power range channel output and Adjust power range channel output if calorimetric exceeds power range channel output by ≥ ~~[~~2~~]~~% RTP. (1)	24 hours	DOC L14 (23) T 4.1-1 Item 3 Col. "Check"
SR 3.3.1.3	NOTE: Not required to be performed until ~~[24]~~ hours after THERMAL POWER is ~~15% RTP~~ ≥ (2) (1) Compare out of core measured AXIAL POWER IMBALANCE (API$_0$) to incore measured AXIAL POWER IMBALANCE (API$_1$) as follows: (RTP/TP)(API$_0$ - API$_1$) = imbalance error Perform CHANNEL CALIBRATION if the absolute value of the imbalance error is ≥ ~~[~~2~~]~~% RTP. (16) INSERT 3.3-3A	31 days	DOC L15 T 4.1-1 Item 4 Col. "Calibrate"
SR 3.3.1.~~4~~5 (2)	Perform CHANNEL FUNCTIONAL TEST.	~~[45]~~ days on a STAGGERED TEST BASIS (1)	T 4.1-1 Column "Test"
SR 3.3.1.~~5~~ 7	NOTE: Neutron detectors are excluded from CHANNEL CALIBRATION. Perform CHANNEL CALIBRATION. INSERT 3.3-3B (41)	DOC L4 31 ~~[92]~~ days (1)	T 4.1-1, Item 4, Column "Calibrate"

(continued)

Not required to be performed until 24 hours after THERMAL POWER is ≥ 15% RTP (41)

INSERT 3.3-3A

and adjust power range channel output if the absolute difference between the power range and incore measurements is ≥ 2% RTP.

INSERT 3.3-3B

Calibrate power range channel output to the calorimetric coincident with the imbalance output being calibrated to the imbalance condition determined by the incore detector system.

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.6	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	~~[18]~~ 18 months
SR 3.3.1.7	-------------------NOTE------------------- Neutron detectors are excluded from RPS RESPONSE TIME testing. -- Verify that RPS RESPONSE TIME is within limits.	[18] months on a STAGGERED TEST BASIS

DOC L20

T 4.1-1 Column "Calibrate"

7

CTS → | 3.5.1.1 | T3.5.1-1 Col. D | T4.1-1 |

CTS

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION ~~A~~.1 (13) (6)	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE (1)
1. Nuclear Overpower -	(24)	(6)	(4)	
a. High Setpoint	1,2(a) 3(d)	~~B~~ C	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 ~~SR 3.3.1.7~~ (7)	≤ ~~[104.9]%~~ 105.5 RTP SR 3.3.1.4 SR 3.3.1.6 (41)
b. Low Setpoint	2(b),3(b) 4(b),5(b)	~~E~~ D	SR 3.3.1.1 ~~SR 3.3.1.5~~ ~~SR 3.3.1.7~~ (7)	≤ 5% RTP SR 3.3.1.6
2. RCS High Outlet Temperature	1,2	~~B~~ C	SR 3.3.1.1 SR 3.3.1.~~4~~ 5 SR 3.3.1.6	≤ ~~[618]~~ °F
3. RCS High Pressure	1,2(a), 3(d) (24)	~~B~~ C	SR 3.3.1.1 SR 3.3.1.~~4~~ 5 SR 3.3.1.6 ~~SR 3.3.1.7~~ (7)	≤ ~~[2355]~~ psig
4. RCS Low Pressure	1,2(a)	~~B~~ C	SR 3.3.1.1 SR 3.3.1.~~4~~ 5 SR 3.3.1.6 ~~SR 3.3.1.7~~ (7)	≥ ~~[1800] psig~~ As specified in the COLR (39)
5. RCS Variable Low Pressure	1,2(a)	~~B~~ C	SR 3.3.1.1 SR 3.3.1.~~4~~ 5 SR 3.3.1.6	≥ ~~([11.59] T_{out} - [5037.8])~~ psig
6. Reactor Building High Pressure	1,2,3(c)	~~B~~ C	SR 3.3.1.1 SR 3.3.1.~~4~~ 5 SR 3.3.1.6	≤ ~~[4]~~ psig (41)
7. Reactor Coolant Pump to Power	1,2(a)	~~B~~ C	SR 3.3.1.1 SR 3.3.1.~~4~~ 5 SR 3.3.1.6 ~~SR 3.3.1.7~~ (7)	~~[5]%~~ >2 RTP with ≤ 2 pumps operating SR 3.3.1.4 (2)
8. Nuclear Overpower ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ Flux/Flow Imbalance (4)	1,2(a)	~~B~~ C	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 ~~SR 3.3.1.7~~ (7)	~~Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE setpoint envelope~~ As specified in the COLR
9. Main Turbine Trip (~~Control Oil~~ Hydraulic Fluid Pressure) (4)	≥ ~~[45]%~~ 30 RTP (46)	~~E~~ E	~~SR 3.3.1.1~~ SR 3.3.1.~~4~~ 5 SR 3.3.1.6	≥ ~~[45]~~ 800 psig (47)
10. Loss of Main Feedwater Pumps (~~Control~~ Hydraulic Oil Pressure) (4)	≥ ~~[15]~~ 2 % RTP	~~F~~ F	~~SR 3.3.1.1~~ SR 3.3.1.~~4~~ 5 SR 3.3.1.6	≥ ~~[55]~~ 75 psig
11. Shutdown Bypass RCS High Pressure	2(b),3(b) 4(b),5(b)	~~E~~ D	SR 3.3.1.1 SR 3.3.1.~~4~~ 5 SR 3.3.1.6	≤ ~~[1720]~~ psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. DOC M2 T2.3-1 Col "Shutdown Bypass"

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. DOC M2

(d) when not in shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. (24) DOC M2

CTS

3.3 INSTRUMENTATION

4

3.3.2 Reactor Protect~~ion~~ ive System (RPS) Manual Reactor Trip

LCO 3.3.2 The RPS Manual Reactor Trip Function shall be OPERABLE. 3.5.1.1 T 3.5.1-1, Col. C, Funct. 3

2

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any ~~CONTROL ROD~~ control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. 3.5.1.1 DOC M3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Manual Reactor Trip Function inoperable.	A.1 Restore Function to OPERABLE status.	1 hour	DOC L3
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. <u>AND</u> B.2 Open all CRD trip breakers.	12 ~~6~~ hours 5 12 ~~6~~ hours	T 3.5.1-1 Funct. 3, Col. D DOC M1
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers.	6 hours	DOC M3

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL FUNCTIONAL TEST.	Once prior to each reactor startup if not performed within the previous 7 days

4.1.1 + T 4.1-1 Item 37 Column "Test"

CTS

3.3 INSTRUMENTATION

4

3.3.3 Reactor Protective ~~Protection~~ System (RPS) – Reactor Trip Module (RTM)

LCO 3.3.3 Four RTMs shall be OPERABLE. 3.5.1.1 T3.5.1-1, Funct 17 Col (C)

2

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any ~~CONTROL ROD~~ drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. 3.5.1.1 Doc M3

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One RTM inoperable.	A.1.1	Trip the associated CRD trip breaker.	1 hour	T3.5.1-1, Col D + Note f.1
	<u>OR</u>			
	A.1.2	Remove power from the associated CRD trip breaker.	1 hour	T3.5.1-1, Col D + Note f.2
	<u>AND</u>			
	A.2	Physically remove the inoperable RTM.	1 hour	DOC M4
B. Two or more RTM's inoperable in MODE 1, 2, or 3. <u>OR</u> Required Action and associated Completion Time not met in MODE 1, 2, or 3. 17	B.1	Be in MODE 3.	12 ~~6~~ hours	DOC M5
	<u>AND</u>			
	B.2.1	Open all CRD trip breakers.	12 ~~6~~ hours 5	
	<u>OR</u>			
	B.2.2	Remove all power from ~~to the~~ CRD trip breakers ~~System~~. 42	12 ~~8~~ hours	

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met in MODE 4 or 5. (17) Two or more RTMs inoperable in MODE 4 or 5. OR	C.1 Open all CRD trip breakers. OR C.2 Remove all power from ~~to the~~ CRD trip breakers ~~System~~. (42)	6 hours Doc M3 6 hours Doc M3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	(43) ~~-------------------NOTE-------------------~~ ~~When an RTM is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided at least two RTM channels are OPERABLE.~~ Perform CHANNEL FUNCTIONAL TEST.	(31) ~~45~~ 31 days (44) ~~on a STAGGERED TEST BASIS~~ 4.1.1, T4.1-1 Item 1 Column "Test"

CTS

3.3 INSTRUMENTATION 2

3.3.4 CONTROL ROD Drive (CRD) Trip Devices

LCO 3.3.4 The following CRD trip devices shall be OPERABLE: 3.5.1.1

a. Two AC CRD trip breakers; T3.5.1-1, Funct. 18+19

b. Two DC CRD trip breaker pairs; and Col C

c. Eight electronic trip assembly (ETA) relays.

APPLICABILITY: MODES 1 and 2, 3.5.1.1
MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal. DOC M3

ACTIONS

NOTE

Separate Condition entry is allowed for each CRD trip device. DOC A6

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. One or ~~more~~ CRD trip breake~~r(s)~~ ~~[~~or breaker pair~~]~~ ~~undervoltage or shunt~~ diverse trip Functions inoperable. 2 25	A.1 1 OR	Trip the CRD trip breaker.	48 hours	T3.5.1-1, COL D + Note i.2
	A.2	Remove power from the CRD trip breaker.	48 hours	DOC L10
B. One or ~~more~~ CRD trip breake~~r(s)~~ ~~[~~or breaker pair~~]~~ inoperable for reasons other than those in Condition A. 2	B.1 1 OR	Trip the CRD trip breaker.	1 hour	DOC L10
	B.2	Remove power from the CRD trip breaker.	1 hour	Table 3.5.1-1, COL D + Note i.1

(continued)

 CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
C. One or more ETA relays inoperable.	C.1	Transfer affected CONTROL ROD group to power supply with OPERABLE ETA relays.	1 hour	DOC L11
	OR			
	C.2	Trip corresponding AC CRD trip breaker. (5)	1 hour (2)	T3.5.1-1, Note (j)
D. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	D.1	Be in MODE 3.	~~6~~ 12 hours	DOC M5
	AND			
	D.2.1	Open all CRD trip breakers.	~~6~~ 12 hours (5)	DOC M5
	OR			
	D.2.2	Remove all power ~~to~~ from ~~the~~ CRD ~~System~~ trip breakers.	~~6~~ 12 hours (42)	DOC M5
E. Required Action and associated Completion Time not met in MODE 4 or 5.	E.1	Open all CRD trip breakers.	6 hours	DOC M3
	OR			
	E.2	Remove all power ~~to~~ from ~~the~~ CRD ~~System~~ trip breakers.	6 hours (42)	DOC M3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	31 days	4.1.1 + T4.1-1, Item 2 Column, "Test"

 CTS

3.3 INSTRUMENTATION

3.3.5 Engineered ~~Safety Feature Actuation~~ Safeguards Protective 4 System ~~(ESFAS)~~ ESPS Analog 4 Instrumentation

LCO 3.3.5 Three channels of ~~ESFAS~~ ESPS 4 analog instrumentation for each Parameter in Table 3.3.5-1 shall be OPERABLE in each ESFAS train. 9

3.5.1.1
T3.5.1-1, Col C
Funct. Units
12a+b, 13 a+b,
14a + 15a

APPLICABILITY: According to Table 3.3.5-1.

3.5.1.1
3.5.3 Note (1)+(2)

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each Parameter.

DOC A6

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one channel inoperable.	A.1 Place channel in trip.	1 hour
B. One or more Parameters with two or more channels inoperable, OR 10 Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2.1 --------NOTE-------- Only required for RCS Pressure – Low ~~setpoint~~. 11 -------------------- Reduce RCS pressure < ~~[1800]~~ 1750 1 psig. AND	~~6~~ 12 5 hours 36 hours (continued)

DOC L5

T3.5.1-1, COL D for Items 12.a, 12.b, 13.a, 13.b, 14.a + 15.a

DOC A8

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2 --------NOTE--------- Only required for RCS Pressure – Low Low ~~setpoint~~. -------------------- Reduce RCS pressure < ~~[~~900~~]~~ psig.	36 hours DOC A8
	AND	
	B.2.3 --------NOTE--------- Only required for Reactor Building Pressure – High ~~setpoint~~ and High High ~~setpoint~~. -------------------- Be in MODE 5.	36 hours T 3.5.1-1 Note (e)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.5.1 Perform CHANNEL CHECK.	12 hours 4.1.1 + T4.1-1, Col "check" Items 15, 17, 19 DOC M21 for Item 21

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
SR 3.3.5.2	-----NOTE----- When an ~~ESFAS~~ analog 4 ESPS 4 channel is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided the remaining two channels of ~~ESFAS~~ ESPS analog 4 instrumentation are OPERABLE or tripped. ------------ Perform CHANNEL FUNCTIONAL TEST.	31 days	DOC L6 4.1.1 + T4.1-1, Col "Test," Items 15,17,19,21
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	~~[18]~~ 18 1 months	4.1.1 + T4.1-1, Col "Calibrate," Items 15,17,19,21
~~SR 3.3.5.4~~	~~Verify ESFAS RESPONSE TIME within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~	7

CTS

Table 3.3.5-1 (page 1 of 1)
Engineered Safeguards Protective ~~Safety Feature Actuation~~ System 4 Analog Instrumentation

3.5.3 Col. Setpoint

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE		
1. Reactor Coolant System Pressure — Low ~~Setpoint (HPI Actuation, RB Isolation, RB Cooling, EDG Start)~~ 11	≥ 1750 ~~[1800]~~ psig	≥ 1500 ~~[1600]~~ psig	1	3.5.3 + Note (1)
2. Reactor Coolant System Pressure — Low Low ~~Setpoint (HPI Actuation, LPI Actuation, RB Isolation, RB Cooling)~~	≥ ~~[900]~~ psig	≥ 500 ~~[400]~~ psig	1	3.5.3 + Note (2)
3. Reactor Building (RB) Pressure — High ~~Setpoint (HPI Actuation, LPI Actuation, RB Isolation, RB Cooling)~~ 11	1,2,3,4	≤ 4 ~~[5]~~ psig	1	3.5.3 + 3.5.1.1
4. Reactor Building Pressure — High High ~~Setpoint (RB Spray Actuation)~~	1,2,3,4	≤ 15 ~~[30]~~ psig	1	3.5.3 + 3.5.1.1

4

3.3 INSTRUMENTATION

3.3.6 Engineered ~~Safety Feature Actuation~~ Safeguards Protective System ~~(ESFAS)~~ ESPS Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the ~~ESFAS~~ ESPS Functions below shall be OPERABLE: 3.5.1.1

14

a. High Pressure Injection; Reactor Building (RB) Non-Essential Isolation, Keowee Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input
(ES Channels 1 and 2) T 3.5.1-1, Col C Funct. Unit 12

b. Low Pressure Injection; and RB Essential Isolation
(ES Channels 3 and 4) T 3.5.1-1, Funct Unit 13

c. ~~Reactor Building (RB)~~ Cooling; and RB Essential Isolation and Penetration Room Ventilation
(ES Channels 5 and 6) T 3.5.1-1 Funct. Unit 14

d. RB Spray.
(ES Channels 7 and 8) T 3.5.1-1 Funct. Unit 15

~~e. RB Isolation; and~~

~~[f. Control Room Isolation.]~~

APPLICABILITY: MODES 1, 2, and 3, 13
MODES 3 and 4 when associated engineered safeguard equipment is required to be OPERABLE. 3.5.1.1

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each Function. DOC A6

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more ~~ESFAS~~ ESPS (4) Functions with one channel inoperable.	A.1	Restore channel to OPERABLE status.	72 hours DOC L7
B. Required Action and associated Completion Time not met.	B.1 AND	Be in MODE 3.	12 ~~6~~ hours (5) T 3.5.1-1 COL D for items 12.c, 13.c, 14.b + 15.b (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Be in MODE 5.	36 hours

T3.5.1-1 Note (e)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.6.1 Perform CHANNEL FUNCTIONAL TEST.	~~[18]~~ months

4.1.1 + T4.1-1, COL "Test" Items 41-48

ESPS - 4 - Digital

3.3 INSTRUMENTATION Safeguard Protective

3.3.7 Engineered ~~Safety Feature Actuation~~ System (~~ESFAS~~) Automatic Actuation Logic Channels

LCO 3.3.7 Eight ~~All the~~ ~~ESFAS~~ Digital automatic actuation logic Channels ~~matrices~~ shall be OPERABLE. — 14 — 4   3.5.1.1 T3.5.1-1, Col C 12d, 13d, 14c, 15c

APPLICABILITY: MODES 1, 2, and 3,
MODES 3 and 4 when associated engineered safeguard (ES) equipment is required to be OPERABLE. — 13   3.5.1.1

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each automatic actuation logic Channel ~~matrix.~~ — 14   DOC A6

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more digital automatic actuation logic Channels ~~matrices~~ inoperable. — 14	A.1	Place associated component(s) in ES ~~engineered safeguard~~ configuration.	1 hour
	OR		
	A.2	Declare the associated component(s) inoperable.	1 hour

T3.5.1-1 Col D 12d, 13d, 14c, 15c — DOC L12

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE 4	FREQUENCY	
SR 3.3.7.1	Perform digital automatic actuation logic CHANNEL FUNCTIONAL TEST.	31 days ~~on a STAGGERED TEST BASIS~~ 15	4.1.1 + T4.1-1 COL "Test" Items 14, 16, 18+20

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3.3 INSTRUMENTATION

3.3.8 Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)

LCO 3.3.8 Three channels of loss of voltage Function and three channels of degraded voltage Function EDG LOPS instrumentation per EDG shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4,
When associated EDG is required to be OPERABLE by LCO 3.8.2 "AC Sources – Shutdown."

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one channel per EDG inoperable.	A.1 Place channel in trip.	1 hour
B. One or more Functions with two or more channels per EDG inoperable.	B.1 Restore all but one channel to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Enter applicable Condition(s) and Required Action for EDG made inoperable by EDG LOPS.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.3.8.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.8.2	-------------------NOTE------------------- When EDG LOPS instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed as follows: (a) up to 4 hours for the degraded voltage Function, and (b) up to 4 hours for the loss of voltage Function, provided the two channels monitoring the Function for the bus are OPERABLE or tripped. --	
	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.8.3	Perform CHANNEL CALIBRATION with setpoint Allowable Value as follows: a. Degraded voltage ≥ [ ] and ≤ [ ] V with a time delay of [ ] seconds ± [ ] seconds at [ ] V; and b. Loss of voltage ≥ [ ] and ≤ [ ] V with a time delay of [ ] seconds ± [ ] seconds at [ ] V.	18 months

3.3 INSTRUMENTATION

3.3.~~17~~ 8 2 Post Accident Monitoring (PAM) Instrumentation

LCO 3.3.~~17~~ 8 2 The PAM instrumentation for each Function in Table 3.3.~~17~~ 8-1 shall be OPERABLE. 2 3.5.6.1 T3.5.6-1, Col. A 3.3.4.a 3.4.1.b

APPLICABILITY: MODES 1, 2, and 3. T3.5.6-1, Col. C 3.3.4 3.4.1

ACTIONS

------NOTE------
Not applicable to Functions 14, 18, 19, and 20.
------ 45

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

1. LCO 3.0.4 is not applicable. DOC A28/M26

2. Separate Condition entry is allowed for each Function. DOC A17/M26

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One or more Functions with one required channel inoperable.	A.1 Restore required channel to OPERABLE status.	30 days	T3.5.6-1, Act. 1-5 Doc M26 Doc M36
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action in accordance with Specification 5.6.8.6 2	Immediately	T3.5.6-1, Act. 3 Doc L22 Doc L23 Doc L24 Doc M26 Doc M36
C. ---------NOTE--------- Not applicable to ~~hydrogen monitor channels~~ Functions 10, 14, 18, 19, and 20 45 --------------------- One or more Functions with two required channels inoperable.	C.1 Restore one channel to OPERABLE status.	7 days	T3.5.6-1, Act 1-5 Doc M26 3.4.3.b

(continued)

--- NOTE ---
Only applicable to Function 10.
45

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
D. Two required ~~hydrogen monitor~~ channels inoperable.	D.1 Restore one required ~~hydrogen monitor~~ channel to OPERABLE status.	72 hours	T3.5.6-1, Act 2
G ~~E~~. Required Action and associated Completion Time of Condition C, D or ~~D~~ E not met.	G ~~E~~.1 Enter the Condition referenced in Table 3.3.~~17~~8-1 for the channel. 2	Immediately	3.5.6.2 T3.5.6-1, Col B DOC M26 DOC A11 DOC M36
H ~~F~~. As required by Required Action G ~~E~~.1 and referenced in Table 3.3.~~17~~8-1. 2	H ~~F~~.1 Be in MODE 3. AND H ~~F~~.2 Be in MODE 4.	12 ~~6~~ hours 5 18 ~~12~~ hours	3.3.4.a(2) T3.5.6-1, Act 1+2 3.4.3.b ~~DOC M26~~ 3.3.4.a.2 DOC M26, M27 T3.5.6-1, Act. 4 3.4.3.b
I ~~G~~. As required by Required Action G ~~E~~.1 and referenced in Table 3.3.~~17~~8-1. 2	I ~~G~~.1 Initiate action in accordance with Specification 5.6.~~8~~6. 2	Immediately	DOC L23 DOC L21 DOC M26

INSERT 3.3-41A 45

INSERT 3.3-41A

CTS

E.	--------NOTE-------- Only applicable to Function 14. ------------------ One required channel inoperable.	E.1 Restore required channel to OPERABLE status.	24 hours	3.3.4.a(2)
F.	--------NOTE-------- Only applicable to Functions 18, 19, and 20. ------------------ One or more Functions with required channel inoperable.	F.1 Declare the affected train inoperable.	Immediately	DOC M26

except where indicated

SURVEILLANCE REQUIREMENTS

NOTE

These SRs apply to each PAM instrumentation Function in Table 3.3.~~17~~ 8-1. 2

DOC A11
4.1.1

SURVEILLANCE	FREQUENCY
SR 3.3.~~17~~ 8.1 2 Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days • T4.1-1, Col. "Check" for Items 5, 26, 30, 39, 49, 55, 56, 60, 61 • DOC M26 • DOC M33
SR 3.3.~~17~~ 8.2 2 NOTE 1. Neutron detectors are excluded from CHANNEL CALIBRATION. Perform CHANNEL CALIBRATION.	~~[18]~~ months 1 • T4.1-1, Col. "Calibrate" for Items 26, 29, 30, 39, 49, 54, 56, 58-61 • M26 • M29

2. Not Applicable to PAM Functions 7 and 10

29

INSERT 3.3-42A

INSERT 3.3-42A

CTS

SR 3.3.8.2	-------------------NOTE------------------- Only applicable to PAM Functions 7 and 10. -- Perform CHANNEL CALIBRATION.	T4.1-1, Col "Calibrate" for Items 55+57 12 months

Table 3.3.~~17~~ 8 -1 2 (page 1 of 1)
Post Accident Monitoring Instrumentation

	FUNCTION	REQUIRED CHANNELS	CONDITIONS REFERENCED FROM REQUIRED ACTION ~~E~~.1 G 2	
1.	Wide Range Neutron Flux	2	~~F~~ H	T 3.5.6-1, Instr. 9
2.	RCS Hot Leg Temperature	2 per loop	~~F~~ H	DOC M26
INSERT B33-43A ~~3.~~	~~RCS Cold Leg Temperature~~	~~2 per loop~~	~~F~~	T 3.5.6-1, Instr. 5
4.	RCS Pressure (Wide Range)	2	~~F~~ H	DOC M26
5.	Reactor Vessel ~~Water~~ Head Level	2	I ~~F~~	T 3.5.6-1, Instr. 6
6.	Containment Sump Water Level (Wide Range)	2	H ~~F~~	T 3.5.6-1, Instr. 2
7.	Containment Pressure (Wide Range)	2	H ~~F~~	T 3.5.6-1, Instr. 1
8.	Containment Isolation Valve Position	2 per penetration flow path(a)(b)(c)	H ~~F~~	DOC M26
9.	Containment Area Radiation (High Range)	2	I ~~F~~	T 3.5.6-1, Instru. 3
10.	Containment Hydrogen Concentration	2	H ~~F~~	T 3.5.6-1, Instr. 4
11.	Pressurizer Level	2	H ~~F~~	DOC M26
INSERT B33-43B 12.	Steam Generator Water Level (upper Surge)	2 per SG	H ~~F~~	DOC M26
~~13.~~ 15	~~Condensate Storage~~ Tank Level	2	H ~~F~~	DOC M26
~~14.~~ 16	Core Exit Temperature	2 independent sets of 5~~(c)~~ (d)	H ~~F~~	T 3.5.6-1, Instr. 7 + Note a
INSERT B3.3-43C ~~15.~~ 21	Emergency Feedwater Flow	2 per SG	H ~~F~~	3.4.1.b

~~NOTE: Table 3.3.17-1 shall be amended for each unit as necessary to list all U.S. NRC Regulatory Guide 1.97, Type A instruments and all U.S. NRC Regulatory Guide 1.97, Category I, non-Type A instruments in accordance with the unit's U.S. NRC Regulatory Guide 1.97, Safety Evaluation Report.~~

(a) Not required for isolation valves whose associated penetration is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured. DOC M26

(b) Only one position indication channel is required for penetration flow paths with only one installed control room indication channel. DOC M26

~~(c)~~ (d) The subcooling margin monitor takes the average of the five highest CETs for each of the ICCM trains. DOC A30

(c) Position indication requirements apply only to containment isolation valves that are electrically controlled. DOC M26

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				CTS
INSERT 3.3-43A				
3.	RCS Hot Leg Level	2	I	T3.5.6-1 Instr. 5
INSERT 3.3-43B				
13.	Steam Generator Pressure	2 per SG	H	Doc M26
14.	Borated Water Storage Tank Water Level	2	H	3.3.4.a
INSERT 3.3-43C				
17.	Subcooling Monitor	2	H	T3.5.6-1 Instr 8
18.	HPI System Flow	1 per train	NA	Doc M26
19.	LPI System Flow	1 per train	NA	Doc M26
20.	Reactor Building Spray Flow	1 per train	NA	Doc M26

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3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux

LCO 3.3.9 Two source range neutron flux channels shall be OPERABLE. 3.5.1.1

T3.5.1-1, COL C Item 2

-----------------------------NOTE-----------------------------
~~High voltage to detector may be de-energized above 1E-10 amp on intermediate range channels.~~
-- 18

APPLICABILITY: MODES 2, 3, 4, and 5. 3.5.1.1 T3.5.1-1, Note b

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required 4 source range neutron flux channel inoperable with THERMAL POWER level ≤ ~~1E-10 amp~~ 4E-4% RTP 22 on the ~~intermediate~~ wide 4 range neutron flux channels.	A.1	Restore channel to OPERABLE status.	Prior to increasing THERMAL POWER DOC M13
B. Two required 4 source range neutron flux channels inoperable with THERMAL POWER level ≤ ~~1E-10 amp~~ on the ~~intermediate~~ wide 4 range neutron flux channels.	B.1	Suspend operations involving positive reactivity changes.	Immediately DOC M13
	<u>AND</u>		
	B.2	Initiate action to insert all CONTROL RODS.	Immediately
	<u>AND</u>		
	B.3	Open CONTROL ROD drive trip breakers.	1 hour
	<u>AND</u>		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.4 Verify SDM ~~is~~ ~~≥ 1% Δk/k~~. to be within the limit ~~provided~~ specified in the COLR (2)	1 hour DOC M13 AND Once per 12 hours thereafter TSTF-009, Rev. 1
C. One or more source range neutron flux channel(s) inoperable with THERMAL POWER level > ~~1E-10 amp~~ 4E-4% RTP (22) on the ~~intermediate~~ wide (4) range neutron flux channels.	C.1 Initiate action to restore affected channel(s) to OPERABLE status.	1 hour T3.5.1-1, Note C

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours 4.1.1, T4.1-1 Col "Check" Item 6
SR 3.3.9.2	-------------------NOTE-------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	~~[18]~~ months (1) DOC M14

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.9.3	Verify at least one decade overlap with intermediate range neutron flux channels.	Once each reactor startup prior to source range counts exceeding 10^5 cps if not performed within the previous 7 days

(20)

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3.3 INSTRUMENTATION

3.3.10 ~~Intermediate~~ wide Range Neutron Flux 4

LCO 3.3.10 Two ~~intermediate~~ wide range neutron flux channels shall be OPERABLE. 3.5.1.1 T 3.5.1-1, Col C Item 1

APPLICABILITY: MODE 2,
~~When~~ with any ~~CONTROL ROD~~ control rod drive (CRD) trip breaker ~~is~~ in the closed position and the CRD System ~~is~~ capable of rod withdrawal MODES 3, 4, and 5 19 3.5.1.1 T 3.5.1-1, Note (6)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One required channel inoperable. 4	A.1 Reduce THERMAL POWER to < ~~1E-10 amp~~ 4E-4% RTP. 22	2 hours	T3.5.1-1 Col D
B. Two required channels inoperable. 4	B.1 Suspend operations involving positive reactivity changes.	Immediately	3.5.1.5
	AND		
	B.2 Open CRD trip breakers.	1 hour	3.5.1.5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.3.10.1 Perform CHANNEL CHECK.	12 hours	4.1.1 T4.1-1 Col "Check" Item 5

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.10.2	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	9 ~~[18]~~ months 1
SR 3.3.10.3	Verify at least one decade overlap ~~with power~~ between source range and wide 4 range neutron flux channels.	Once each reactor startup prior to ~~the~~ source 4 ~~intermediate~~ range indication exceeding ~~1E-6 amp~~ 10^5 cps if not performed within the previous 7 days

DOC M14

3.5.1.5

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MSLB Detection and MFW Isolation

CTS

3.3 INSTRUMENTATION

Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation

3.3.11 ~~Emergency Feedwater Initiation and Control (EFIC) System~~ Instrumentation

T3.5.1-1, Item 20, Col. C, 3.5.7.1

LCO 3.3.11 ~~The EFIC System instrumentation channels for each Function in Table 3.3.11-1~~ Three MSLB Detection and MFW Isolation instrumentation channels per steam generator (SG) shall be OPERABLE.

APPLICABILITY: ~~According to Table 3.3.11-1~~ MODES 1 and 2, MODE 3 with main steam header pressure ≥ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed

3.5.7 Applic.

ACTIONS

-------------------------------------NOTE-------------------------------------
Separate Condition entry is allowed for each ~~Function~~ SG (MFW Isolation Function)

DOC A27

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~Emergency Feedwater (EFW) Initiation, Main Steam Line Isolation, or Main Feedwater (MFW)~~ Isolation Functions ~~listed in Table 3.3.11-1~~ with one channel inoperable.	A.1 Place channel(s) in ~~bypass or~~ trip. ~~AND~~ ~~A.2 Place channel(s) in trip.~~	~~1~~ 4 hours ~~72 hours~~
~~B. One or more EFW Initiation, Main Steam Line Isolation, or MFW Isolation Functions listed in Table 3.3.11-1 with two channels inoperable.~~	~~B.1 Place one channel in bypass.~~ ~~AND~~ ~~B.2 Place second channel in trip.~~ ~~AND~~	~~1 hour~~ ~~1 hour~~

T 3.5.1, Note (K)

(continued)

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ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. (continued)	B.3	Restore one channel to OPERABLE status.	72 hours
C. One EFW Vector Valve Control channel inoperable.	C.1	Restore channel to OPERABLE status.	72 hours
D. Required Action and associated Completion Time not met for Functions ~~1.a or 1.b~~.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2.1	--------NOTE--------- Only required for Function 1a. --------------------- Open CONTROL ROD drive trip breakers.	6 hours
	AND		
	D.2.2	--------NOTE--------- Only required for Function 1b. --------------------- Be in MODE 4.	12 hours
E. Required Action and associated Completion Time not met for Function 1.d.	E.1	Reduce THERMAL POWER to ≤ 10% RTP.	6 hours

(continued)

50 28 <Except as marked>

MSLB Detection and MFW Isolation

CTS

One or more MFW Isolation Functions with two or more channels inoperable
OR

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B ~~F~~. Required Action and associated Completion Time not met ~~for Functions 1.c, 2, 3 or 4~~.	B.1 Be in MODE 3. AND B.2.1 ~~F.1~~ Reduce ~~once through steam generator~~ main steam header pressure to < ~~750~~ 700 psig. OR B.2.2 Close all MFCVs and SFCVs.	12 hours 5 18 ~~12~~ hours 5 18 hours

T3.5.1-1 Item 20, Col D

DOC L18

SURVEILLANCE REQUIREMENTS

~~NOTE~~
~~Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.~~

SURVEILLANCE	FREQUENCY
SR 3.3.11.1 Perform CHANNEL CHECK.	12 hours
~~SR 3.3.11.2 Perform CHANNEL FUNCTIONAL TEST.~~	~~31 days~~
SR 3.3.11.2 ~~3~~ Perform CHANNEL CALIBRATION.	~~[~~18~~]~~ months
~~SR 3.3.11.4 Verify EFIC RESPONSE TIME is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~

4.1.1 T4.1-1 Item 62 Col "Check"

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4.1.1 T4.1-1 Item 62 Col "Calibrate"

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Table 3.3.11-1 (page 1 of 1)
Emergency Feedwater Initiation and Control System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. EFW Initiation				
a. Loss of MFW Pumps (Control Oil Pressure)	1,2(a),3(a)	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	> [55] psig
b. SG Level - Low	1,2,3	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3 SR 3.3.11.4	≥ [9] inches
c. SG Pressure - Low	1,2,3(b)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ [600] psig
d. RCP Status	≥ 10% RTP	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	NA
2. EFW Vector Valve Control				
a. SG Pressure - Low	1,2,3(b)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≥ [600] psig
b. SG Differential Pressure - High	1,2,3(b)	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≤ [125] psid
[c. SG Level - High	1,2,3(b)	4	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3	≤ [ ] inches]
3. Main Steam Line Isolation				
a. SG Pressure - Low	1,2,3(b)(c)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3 SR 3.3.11.4	≥ [600] psig
4. MFW Isolation				
a. SG Pressure - Low	1,2,3(b)(d)	4 per SG	SR 3.3.11.1 SR 3.3.11.2 SR 3.3.11.3 SR 3.3.11.4	≥ [600] psig

(a) When not in shutdown bypass.

(b) When SG pressure ≥ 750 psig.

(c) Except when all associated valves are closed and [deactivated].

(d) Except when all [MFSVs], [MFCVs], [or associated SFCVs] are closed and [deactivated] [or isolated by a closed manual valve].

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CTS

Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation

3.3 INSTRUMENTATION

3.3.12 ~~Emergency Feedwater Initiation and Control (EFIC)~~ Manual Initiation

LCO 3.3.12 Two manual initiation switches ~~per actuation channel for each of the following EFIC Functions~~ shall be OPERABLE~~:~~. 3.5.7.1 T3.5.1-1, Item 22, Col. C

~~a. Steam generator (SG) A Main Feedwater (MFW) Isolation;~~

~~b. SG B MFW Isolation;~~

~~c. SG A Main Steam Line Isolation;~~

~~d. SG B Main Steam Line Isolation; and~~

~~e. Emergency Feedwater Actuation.~~

APPLICABILITY: MODES 1~~,~~ and 2, ~~and 3.~~
MODE 3 with main steam header pressure ≥ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed. 3.5.7 Applic.

ACTIONS

~~-------------------------------------NOTE-------------------------------------~~
~~Separate Condition entry is allowed for each Function.~~
~~--~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One or more EFIC Function(s) with one or both manual initiation switches inoperable in one actuation channel.~~	~~A.1 Place actuation channel for the associated EFIC Function(s) in trip.~~	~~72 hours~~
A ~~B~~. ~~One or more EFIC Function(s) with one or both~~ manual initiation switch~~es~~ inoperable ~~in both actuation channels~~.	A ~~B~~.1 ~~Restore one actuation channel for the associated EFIC Function(s) to OPERABLE status~~. Restore manual initiation switch to OPERABLE status.	72 hours ~~1 hour~~ T3.5.1-1, Note (l)

(continued)

MSLB Detection and MFW Isolation

CTS

~~EFIC~~ Manual Initiation
3.3.12

50

28 <Except as marked>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~C~~ B. Two manual initiation switches inoperable OR Required Action and associated Completion Time of Condition A not met.	~~C~~ B.1 Be in MODE 3. AND ~~C~~ B.2.1 ~~Be in MODE 4.~~ Reduce main steam header pressure to < 700 psig. OR B.2.2 Close all MFCVs and SFCVs.	~~6~~ 12 hours ~~12~~ 18 hours 18 hours

T3.5.1-1 Item 22, Col. D

5

DOC L18

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.12.1 Perform CHANNEL FUNCTIONAL TEST.	18 months ~~31 days~~

4.1.1

T4.1-1 Item 63 Col. "Test"

28 <Except as marked> MSLB Detection and MFW Isolation

~~EFIC~~ Logic
3.3.13

CTS

Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation

3.3 INSTRUMENTATION

3.3.13 ~~Emergency Feedwater Initiation and Control (EFIC)~~ Logic Channels

LCO 3.3.13 Two ~~Channels A and B of each~~ logic ~~Function shown below~~ channels shall be OPERABLE~~:~~. 3.5.7.1 T3.5.1-1, Item 21, Col. C

~~a. Main Feedwater Isolation;~~

~~b. Main Steam Line Isolation;~~

~~c. Emergency Feedwater Actuation; and~~

~~d. Vector Valve Enable Logic.~~

APPLICABILITY: MODES 1, 2, and ~~and 3.~~ MODE 3 with main steam header pressure ≥ 700 psig except when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed. 3.5.7 Applic

ACTIONS

~~NOTE~~
~~Separate Condition entry is allowed for each Function.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~or more channel A Functions inoperable with all channel B Functions OPERABLE; or one or more channel B Functions inoperable with all channel A Functions OPERABLE.~~ logic channel inoperable	A.1 Restore ~~affected~~ inoperable channel to OPERABLE status.	72 hours	T3.5.1-1, Note (l)
B. Required Action and associated Completion Time of Condition A not met. OR Two logic channels inoperable.	B.1 Be in MODE 3. Reduce main steam header pressure to < 700 psig. AND B.2.1 ~~Be in MODE 4.~~ OR B.2.2 Close all MFCVs and SFCVs.	~~6~~ 12 hours 5 ~~12~~ 18 hours 18 hours	T3.5.1-1, Item 21, Col. D — DOC L18

50

(28) <Entire page>

MSLB Detection and MFW Isolation

~~EFIC~~ Logic Channels
3.3.13

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.13.1 Perform CHANNEL FUNCTIONAL TEST.	18 months ~~31 days~~

4.1.1
T4.1-1
Item 63
Col "Test"

3.3 INSTRUMENTATION

3.3.14 Emergency Feedwater Initiation and Control (EFIC)- Emergency Feedwater (EFW) – Vector Valve Logic

LCO 3.3.14 Four channels of the vector valve logic shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One vector valve logic channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.14.1 Perform a CHANNEL FUNCTIONAL TEST.	31 days

28 < Except as marked >

CTS

EFW Pump Initiation Circuitry
~~EFIC System~~ ~~Instrumentation~~
3.3.~~11~~14

3.3 INSTRUMENTATION

3.3.~~11~~14 Emergency Feedwater (EFW) Pump Initiation Circuitry ~~and Control (EFIC) System~~ ~~Instrumentation~~

LCO 3.3.~~11~~14 Two Loss of Main Feedwater (LOMF) pump ~~The EFIC System~~ instrumentation channels for each automatic initiation circuits and ~~Function~~ ~~in Table 3.3.11-1~~ an automatic and manual initiation circuit for each EFW pump shall be OPERABLE. 3.4.1.a 3.4.2

INSERT 3.3-27A

APPLICABILITY: ~~According to Table 3.3.11-1.~~ MODES 1, 2 and 3, MODE 4 when the steam generator is relied upon for heat removal 3.4.1

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each ~~Function.~~ EFW pump initiation circuit. DOC L16

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~Emergency Feedwater~~ ~~(EFW)~~ pump LOMF Initiation ~~, Main Steam Line Isolation, or Main Feedwater (MFW) Isolation Functions listed in Table 3.3.11-1~~ with one circuits channel inoperable.	A.1 Place channel(s) in ~~bypass or~~ trip. ~~AND~~ ~~A.2 Place channel(s) in trip.~~	1 hour ~~72 hours~~
~~B. One or more EFW Initiation, Main Steam Line Isolation, or MFW Isolation Functions listed in Table 3.3.11-1 with two channels inoperable.~~	~~B.1 Place one channel in bypass.~~ ~~AND~~ ~~B.2 Place second channel in trip.~~ ~~AND~~	~~1 hour~~ ~~1 hour~~

~~(continued)~~

B One or more required EFW pump initiation circuits inoperable. OR Required Action and associated Completion Time not met.	B.1 Declare the affected EFW pump(s) inoperable.	Immediately

DOC L16

DOC A10

INSERT 3.3-27A

CTS

-------------------------------NOTE--------------------------------

The EFW pump automatic initiation circuit is not required to be OPERABLE in MODES 3 and 4.

DOC A10 3.4.3.e

----

28 < Except as marked >

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met for Functions 1.c, 2, 3, or 4.	F.1 Reduce once through steam generator pressure to < [7]50 psig.	12 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.

SURVEILLANCE	FREQUENCY
SR 3.3.11.1 Perform CHANNEL CHECK.	12 hours
SR 3.3.~~11.2~~14.1 Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.~~11.3~~14.2 Perform CHANNEL CALIBRATION.	[18] months
SR 3.3.11.4 Verify EFIC RESPONSE TIME is within limits.	[18] months on a STAGGERED TEST BASIS

40

4.1.1
T 4.1-1
Item 53
Col. "Test"

4.1.1
T4.1-1
Item 53
Col. "Calibrate"

7

3.3 INSTRUMENTATION

Turbine Stop Valve (TSV) Closure

3.3.~~11~~ 15 ~~Emergency Feedwater Initiation and Control (EFIC) System Instrumentation~~

LCO 3.3.~~11~~ 15 ~~The EFIC System instrumentation channels for each Function in Table 3.3.11-1~~ Two TSV Closure channels shall be OPERABLE. 3.5.1.1 T 3.5.1-1 Item 16 Col C

APPLICABILITY: ~~According to Table 3.3.11-1.~~ 3.5.1.1

MODES 1, 2 and 3 except when all TSVs are closed

ACTIONS

~~-----NOTE-----~~
~~Separate Condition entry is allowed for each Function.~~

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ~~Emergency Feedwater (EFW) Initiation, Main Steam Line Isolation, or Main Feedwater (MFW) Isolation Functions listed in Table 3.3.11-1 with one~~ TSV Closure channel(s) inoperable.	A.1 ~~Place channel(s) in bypass or trip.~~ Declare the TSVs inoperable	1 hour DOC L19
	~~AND~~	
	~~A.2 Place channel(s) in trip.~~	~~72 hours~~
~~B. One or more EFW Initiation, Main Steam Line Isolation, or MFW Isolation Functions listed in Table 3.3.11-1 with two channels inoperable.~~	~~B.1 Place one channel in bypass.~~	~~1 hour~~
	~~AND~~	
	~~B.2 Place second channel in trip.~~	~~1 hour~~
	~~AND~~	

~~(continued)~~

28 <Entire Page>

TSV Closure

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Required Action and associated Completion Time not met for Functions 1, 2, 3, or 4.	F.1 Reduce once through steam generator pressure to < 750 psig.	12 hours

SURVEILLANCE REQUIREMENTS

NOTE

Refer to Table 3.3.11-1 to determine which SRs shall be performed for each EFIC Function.

SURVEILLANCE	FREQUENCY
SR 3.3.11.1 Perform CHANNEL CHECK.	~~12~~ hours
SR 3.3.~~11~~15.~~2~~1 Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.11.3 Perform CHANNEL CALIBRATION.	[18] months
SR 3.3.11.4 Verify EFIC RESPONSE TIME is within limits.	[18] months on a STAGGERED TEST BASIS

DOC M23

CTS

3.3 INSTRUMENTATION

3.3.~~15~~ 16 Reactor Building (RB) Purge Isolation – High Radiation

LCO 3.3.~~15~~ 16 (2) ~~[One]~~ channel of Reactor Building Purge Isolation – High (1) Radiation shall be OPERABLE. 3.8.10

APPLICABILITY: ~~MODES 1, 2, 3, and 4,~~ (36)
During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within the ~~RB~~ containment (4) 3.8.10

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One channel inoperable in MODE 1, 2, 3, or 4.~~ (36)	~~A.1 Place and maintain RB purge valves in closed positions.~~	~~1 hour~~
~~B. Required Action and associated Completion Time of Condition A not met.~~	~~B.1 Be in MODE 3.~~ ~~AND~~ ~~B.2 Be in MODE 5.~~	~~6 hours~~ ~~36 hours~~
~~C.~~ A. (2) One channel inoperable ~~during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB.~~ (36)	~~C.1~~ A.1 (2) Place and maintain RB purge valves in closed positions. OR (2) ~~C.2.1~~ A.2.1 Suspend CORE ALTERATIONS. AND	Immediately DOC M18 Immediately DOC M18 (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A ~~C~~ 2 (continued)	A ~~C~~.2.2 2 Suspend movement of irradiated fuel assemblies within the ~~RB~~ containment 4	Immediately DOC M18

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.~~15~~ 16.1 2 Perform CHANNEL CHECK.	12 hours DOC M19
SR 3.3.~~15~~ 16.2 2 Perform CHANNEL FUNCTIONAL TEST.	~~92 days~~ 3.8.10, T 4.1-2, Item 4 Once each refueling outage prior to Core ALTERATIONS, or movement of irradiated fuel assemblies within containment 32
SR 3.3.~~15~~ 16.3 2 Perform CHANNEL CALIBRATION ~~with setpoint Allowable Value ≤ [25] mR/hr.~~ 37	~~[18]~~ months 1 DOC M19

35

3.3 INSTRUMENTATION

3.3.16 Control Room Isolation—High Radiation

LCO 3.3.16 [One] channel of Control Room Isolation—High Radiation shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One channel inoperable in MODE 1, 2, 3, or 4.	A.1	--------NOTE--------- Place in toxic gas protection mode if automatic transfer to toxic gas protection mode is inoperable. --------------------	
		Place one OPERABLE Control Room Emergency Ventilation System (CREVS) train in the emergency recirculation mode.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One channel inoperable during [CORE ALTERATIONS or] during movement of irradiated fuel.	C.1	Place one OPERABLE CREVS train in emergency recirculation mode.	Immediately
	OR		
	[C.2.1	Suspend CORE ALTERATIONS.	Immediately]
	AND		
	C.2[.2]	Suspend movement of irradiated fuel assemblies.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.16.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.16.2	-------------------NOTE------------------- When the Control Room Isolation—High Radiation instrumentation is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 3 hours. -- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.16.3	Perform CHANNEL CALIBRATION with setpoint Allowable Value ≤ [25] mR/hr.	[18] months

3.3 INSTRUMENTATION

3.3.18 Remote Shutdown System

LCO 3.3.18 The Remote Shutdown System Functions in Table 3.3.18-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

---NOTES---

1. LCO 3.0.4 is not applicable.
2. Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required Functions inoperable.	A.1 Restore required Function to OPERABLE status.	30 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours
	AND	
	B.2 Be in MODE 4.	12 hours

38

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
[SR 3.3.18.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days]
SR 3.3.18.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	[18] months
SR 3.3.18.3	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION for each required instrumentation channel.	[18] months

Table 3.3.18-1 (page 1 of 1)
Remote Shutdown System Instrumentation and Controls

---NOTE---
This Table is for illustration purposes only. It does not attempt to encompass every Function used at every unit, but does contain the types of Functions commonly found.

FUNCTION/INSTRUMENT OR CONTROL PARAMETER	REQUIRED NUMBER OF FUNCTIONS
1. Reactivity Control	
a. Log Power Neutron Flux	[1]
b. Source Range Neutron Flux	[1]
c. Reactor Trip Circuit Breaker Position	[1 per trip breaker]
d. Manual Reactor Trip	[1]
2. Reactor Coolant System (RCS) Pressure Control	
a. Pressurizer Pressure or RCS Wide Range Pressure	[1]
b. Pressurizer Power Operated Relief Valve (PORV) Control and Block Valve Control	[1]
3. Decay Heat Removal via Steam Generators (SGs)	
a. Reactor Coolant Hot Leg Temperature	[1 per loop]
b. Reactor Coolant Cold Leg Temperature	[1 per loop]
c. Condensate Storage Tank Level	[1]
d. SG Pressure	[1 per SG]
e. SG Level or Emergency Feedwater (EFW) Flow	[1 per SG]
f. EFW Controls	[1]
4. RCS Inventory Control	
a. Pressurizer Level	[1]
b. Reactor Coolant Injection Pump Controls	[1]

33 < For INSERTS 3.3.17 through 3.3.24 except as marked >

3.3 INSTRUMENTATION

3.3.17 Emergency Power Switching Logic (EPSL) Automatic Transfer Function

LCO 3.3.17 Two channels of the EPSL Automatic Transfer Function shall be OPERABLE. TS 3.7.3

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Restore channel to OPERABLE status.	24 hours Act A
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours Act B 84 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.17.1 Perform CHANNEL FUNCTIONAL TEST.	18 months SR 3.7.3.1 SR 3.7.1.14

3.3 INSTRUMENTATION

CTS

3.3.18 Emergency Power Switching Logic (EPSL) Voltage Sensing Circuits

LCO 3.3.18 Three channels of each of the following EPSL voltage sensing circuits shall be OPERABLE: TS 3.7.4

a. Startup Transformer;
b. Standby Bus 1;
c. Standby Bus 2; and
d. Auxiliary Transformer.

---------------------------NOTE---------------------------

1. If both N breakers are open, Auxiliary Transformer voltage sensing circuits are not required to be OPERABLE. LCO Note

2. When not in MODES 1, 2, 3, and 4, only EPSL voltage sensing circuit(s) associated with required AC power source(s) are required to be OPERABLE. DOC M34

APPLICABILITY: MODES 1, 2, 3, 4, 5 and 6, During movement of irradiated fuel assemblies. 49 Applic

ACTIONS

---------------------------NOTE---------------------------

Separate Condition entry is allowed for each circuit. ACT Note

----

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required circuits with one channel inoperable.	A.1 Restore channel to OPERABLE status.	24 hours

ACT A

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met in MODES 1, 2, 3, and 4. 49	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours ACT B 84 hours
C. Two or more channels of a required circuit inoperable when not in MODES 1, 2, 3, and 4. OR Required Action and associated Completion Time not met when not in MODES 1, 2, 3, and 4.	C.1 Declare affected AC power source(s) inoperable.	Immediately DOC M34 49
D. Required Action and associated Completion Time not met during movement of irradiated fuel assemblies.	D.1 Suspend movement of irradiated fuel assemblies.	Immediately DOC M34

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.18.1 Perform CHANNEL FUNCTIONAL TEST.	18 months SR 3.7.4.1

3.3 INSTRUMENTATION

CTS

3.3.19 Emergency Power Switching Logic (EPSL) 230 kV Switchyard Degraded Grid Voltage Protection (DGVP)

LCO 3.3.19 Three DGVP voltage sensing channels and two DGVP actuation logic channels shall be OPERABLE. TS 3.7.6

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One voltage sensing channel inoperable.	A.1 Place channel in trip.	72 hours ACT A
B. One actuation logic channel inoperable.	B.1 Restore channel to OPERABLE status.	72 hours ACT B
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	12 hours ACT C 84 hours
D. Two or more voltage sensing channels inoperable. OR Two actuation logic channels inoperable.	D.1 Declare the overhead emergency power path inoperable.	Immediately ACT D

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.19.1	Perform a CHANNEL FUNCTIONAL TEST.	18 months SR 3.7.6.2
SR 3.3.19.2	Perform a CHANNEL CALIBRATION of the voltage sensing channel with the setpoint allowable value as follows: Degraded voltage ≥ 219 kV and ≤ 222 kV with a time delay of 9 seconds ±1 second.	18 months SR 3.7.6.1

3.3 INSTRUMENTATION

CTS

3.3.20 Emergency Power Switching Logic (EPSL) CT-5 Degraded Grid Voltage Protection (DGVP)

LCO 3.3.20 Three CT-5 DGVP voltage sensing channels and two CT-5 DGVP actuation logic channels shall be OPERABLE. TS 3.7.7

APPLICABILITY: MODES 1, 2, 3, and 4 when the Central Switchyard is energizing the standby buses. Applic

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One voltage sensing channel inoperable.	A.1 Place channel in trip.	72 hours ACT A
B. One actuation logic channel inoperable.	B.1 Restore channel to OPERABLE status.	72 hours ACT B
C. Two or more voltage sensing channels inoperable. OR Two actuation logic channels inoperable. OR Required Action and associated Completion Time of Condition A or B not met.	C.1 Open SL breakers.	1 hour ACT C

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.3.20.1	Perform a CHANNEL FUNCTIONAL TEST.	18 months	SR 3.7.7.2
SR 3.3.20.2	Perform a CHANNEL CALIBRATION of the voltage sensing channel with the setpoint allowable value as follows: a. Degraded voltage ≥ 4143 V and ≤ 4185 V with a time delay of 9 seconds ±1 second for the first level undervoltage inputs; and b. Degraded voltage ≥ 3871 V and ≤ 3901 V for the second level undervoltage inputs.	18 months	SR 3.7.7.1

3.3 INSTRUMENTATION

CTS

3.3.21 Emergency Power Switching Logic (EPSL) Keowee Emergency Start Function

LCO 3.3.21 Two channels of the EPSL Keowee Emergency Start Function shall be OPERABLE. TS 3.7.5

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours	ACT A
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	12 hours 84 hours	ACT B
C. Two channels inoperable.	C.1 Declare both Keowee Hydro Units inoperable.	Immediately	ACT C

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.3.21.1 Perform CHANNEL FUNCTIONAL TEST.	18 months	SR 3.7.5.1 SR 3.7.1.14

(48) <entire page>

3.3 INSTRUMENTATION

3.3.22 Emergency Power Switching Logic (EPSL) Manual Keowee Emergency Start Function

LCO 3.3.22 One channel of the EPSL Manual Keowee Emergency Start Function shall be OPERABLE. DOC M38

APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies. DOC M38

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	A.1 Declare both Keowee Hydro Units inoperable.	Immediately DOC M38

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.22.1 Perform CHANNEL FUNCTIONAL TEST.	12 months DOC M38

TECHNICAL SPECIFICATIONS

NOTE: The first four justifications for these changes from NUREG-1430 were generically used throughout the individual LCO section markups. Not all generic justifications are used in each section.

1 The brackets are removed and the proper plant specific information or value is provided.

2 Editorial changes made for clarity, preference or consistency with the Improved Technical Specifications (ITS) Writer's Guide.

3 The requirement/statement is deleted since it is not applicable to this facility. The following requirements are renumbered, where applicable, to reflect this deletion.

4 Changes are made (additions, deletions, and/or changes to the NUREG) to reflect the facility specific nomenclature, number, reference, system description, or analysis description.

5 The current licensing basis (CLB) permits 12 hours to place a unit in Hot Shutdown (equivalent to ITS MODE 3) when an LCO is not met. To maintain consistency with current procedures, training and staffing requirements, the 12 hours permitted to place a unit in MODE 3 is retained in the ITS.

As a result of modifying the Completion Time to place a unit in MODE 3, Completion Times for concurrent Required Actions within an ACTION were extended from 6 hours to 12 hours. NUREG 3.3.1 Required Action C.2, NUREG 3.3.2 Required Action B.2, NUREG 3.3.3 Required Action B.2.1, and NUREG 3.3.4 Required Action D.2.1 are modified to allow 12 hours to open all control rod drive (CRD) trip breakers consistent with the concurrent Completion Time to place the unit in MODE 3. NUREG 3.3.3 Required Action B.2.2 and NUREG 3.3.4 Required Action D.2.2 are modified to allow 12 hours to remove power from all CRD trip breakers consistent with the concurrent Completion Time to place the unit in MODE 3. Also, NUREG 3.3.1 Required Action G.1 Completion Time is modified to allow 12 hours to reduce power to < 2%. This is necessary since the ONS ITS power reduction is to < 2% versus the NUREG power reduction to < 15%. A power level of < 2% is a condition very close to MODE 3.

6 CTS Table 3.5.1-1 Column C permits unlimited operation with three of the four available RPS instrumentation channels OPERABLE. NUREG LCO 3.3.1 and associated Actions are modified to reflect the CTS provision. NUREG LCO 3.3.1, ACTION A is deleted (and subsequent ACTIONS relettered) since the Required Action only applies when 4 channels are required OPERABLE. Condition B is modified to apply when one required channel is inoperable (one of the three required versus two of the four available) and Required Action B.2 is deleted since this applies to the 4 channel configuration.

7 NUREG SR 3.3.1.7, SR 3.3.5.4 and SR 3.3.11.4 are not adopted. Consistent with current licensing basis, response time testing of RPS, ESPS, and EFIC (ONS equivalent) circuitry is not performed. Plant equipment does not readily lend itself to such testing.

8 Not used.

9 The unit specific design of the ONS ESPS provides for three analog channels of instrumentation for each of the monitored parameters. These three channels provide the required input to each of the eight automatic actuation logic channels. Contrary to the system design depicted in the requirements of the NUREG, these three instrument channels provide input to both trains of automatic actuation logic channels. This unit specific design difference required the deletion of the phrase "in each ESFAS train" from NUREG LCO 3.3.5 as well as appropriate changes to the Bases.

10 NUREG Specification 3.3.5 Condition B has been revised to specify that this Condition applies when two or more channels are inoperable for each of one or more Parameters. This change was made to maintain requirements consistent with CTS Table 3.5.1-1 Column D and Note (e) which provide specific requirements for the inoperability of more than one channel.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if more than one channel is inoperable for each of one or more Parameters. Entry into the Required Actions of ITS 3.3.5 Condition B rather than the ACTION requirements of LCO 3.0.3 is more appropriate because, specific Required Actions, which result in the unit exiting the Applicability for each ESPS instrumentation Parameter, are provided in ITS 3.3.5. These Required Actions consistently result in the unit exiting the specific Applicability within a specific Completion Time. For example, ITS LCO 3.0.3 ACTION requirements would not provide a specific Completion Time for reducing RCS pressure to less than 1750 psig, in the event more than one channel of the RCS Pressure-Low Setpoint Parameter was inoperable, but the required Completion Time would not be specifically identified. [ONS-019]

11 NUREG Table 3.3.5-1 lists equipment actuated by each of the ESPS Parameters. This list is incomplete and has not been included in the ITS in favor of the more complete list provided in the Bases. Also, the term "Setpoint" is removed from the Parameter title in the Table and from Required Actions B.2.1, B.2.2 and B.2.3 since the setpoint is not a parameter. Referring to a setpoint as a parameter is inconsistent with the identification of other parameters and functions throughout the NUREG. Removal of this information represents no actual change in requirements.

12 In the conversion to ITS, NUREG Table 3.3.17-1, Post Accident Monitoring Instrumentation, is modified by Note c which indicates that the

containment isolation valve position indication requirements apply only to containment isolation valves that are electrically controlled. This is consistent with ONS Regulatory Guide 1.97 response for CIV position indication and the NRC's Safety Evaluation Report for this response.

13 The Applicability of NUREG LCO 3.3.6 and LCO 3.3.7 is modified to only include the portions of MODE 3 in which the associated ES equipment is required to be OPERABLE. This change was made to reflect the fact that some ESPS actuated equipment is not required in either MODE 3 or MODE 4. For example, neither CTS nor the proposed ITS requires the High Pressure Injection (HPI) System, which is actuated by the ESPS, to be OPERABLE in MODE 3 with Reactor Coolant Temperature $\leq$ 350°F. This change was made to provide Applicabilities for the ESPS requirements which are consistent with the Applicabilities of the actuated equipment. [ONS-011]

14 The Functions specified in NUREG LCO 3.3.6 are modified to match the Functions as presented in the CTS, UFSAR, and other design basis documents. NUREG Specification 3.3.7 has also been modified to include ONS unit specific terminology. These changes were made to provide requirements consistent with the design of the ONS and consistent with the specific terminology and names associated with the ONS ESPS.

15 The Frequency of SR 3.3.7.1 has been changed to 31 days. The NUREG Frequency of 31 days on a STAGGERED TEST BASIS is not consistent with the CTS. The CTS requires this testing monthly, which is considered administratively equivalent to the proposed 31 day Frequency.

16 The specific details of performance of ITS SR 3.3.1.3 have been removed. These details provided methodology and acceptance criteria not contained in CTS. The removal of these details maintains requirements consistent with CTS. The details of this testing are currently contained in implementing procedures and will be retained there. This change neither adds any new requirements nor removes any existing requirement. In addition, the requirement to perform a CHANNEL CALIBRATION if the absolute value of the imbalance error is $\geq$ [2]% RTP is not included since the Frequency for ITS SR 3.3.1.4 (NUREG SR 3.3.1.5) is the same and this SR also requires the power range channel calibration.

17 NUREG Specification 3.3.3 Conditions B and C are revised to specify that these Conditions also apply when more than one RPS Reactor Trip Module (RTM) is inoperable. This change is made to provide ACTION requirements which specifically remove the unit from the Applicability for this Specification.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if more than one RTM is inoperable. Entry into the Required Actions of ITS 3.3.3 Condition B or C (depending upon the current MODE), rather than the ACTION requirements of LCO 3.0.3, is more

appropriate because specific Required Actions which result in the unit exiting the Applicability for LCO 3.3.3 are provided. These Required Actions result in the unit exiting the specific Applicability by either opening the Control Rod Drive (CRD) trip breakers or removing power from all CRD trip breakers within a specific Completion Time. ITS LCO 3.0.3 ACTION requirements would not require opening the CRD trip breakers or removing power from the CRD system, and therefore, would not result in exiting the Applicability of ITS 3.3.3. [ONS-019]

18 At ONS, the source range detectors are not turned off at power because the wide range instrument channels use one of the same fission chambers that supplies the source range instrument channels. To monitor the source range, two fission chambers are used and the outputs are added together. Only one fission chamber is used for the wide range output.

19 NUREG 3.3.10 Applicability is changed to specify that the wide range instrument channel is required in MODE 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The addition of "MODES 3, 4, and 5" to the second statement of the Applicability was made to maintain the CTS allowance provided by Table 3.5.1-1 Note (b). This Note defines the upper limit of the applicable MODES for the required wide range instrument channel as being 10% indicated neutron power. Without the addition of the appropriate MODES, to the second statement of the Applicability for ITS 3.3.10, a wide range channel would be required at all times in MODE 1. This requirement is inconsistent with the RPS design requirement of the wide range instrument channels which is to provide indication of neutron power while operating at low power levels (MODE 2). The required indication of neutron power level is provided by the power range instruments while in MODE 1.

20 NUREG SR 3.3.9.3 is moved to ITS 3.3.10 as SR 3.3.10.3. This SR provides verification that at least one decade of overlap exists between the source range and wide range instruments when the wide range instruments come on scale. By associating this SR with the LCO for the source range instruments, the NUREG inappropriately establishes the successful performance of this SR as an OPERABILITY requirement for the source range instruments. By associating this SR with LCO 3.3.10, the ONS ITS more appropriately establishes the successful performance of the SR as an OPERABILITY requirement for the wide range instrument channels. This is consistent with CTS 3.5.1.5.

The requirement to verify one decade overlap between the source range and wide range instrument channels ensures a continuous source of power indication is maintained during the approach to criticality. Provided the source range instruments are maintained on scale, a continuous indication of power is maintained, even if the wide range instruments fail to come on scale within the required one decade overlap. By associating this SR with the wide range instrument channels rather than the source range instrument channels, successful performance of this SR

will ensure that the wide range instrument channels are OPERABLE prior to relying upon them as the primary indication of core reactor power. [ONS-021]

21 NUREG SR 3.3.10.3 is not retained in the ITS because no similar requirement to perform this verification exists in the ONS CTS. According to the BASES for SR 3.3.10.3, this SR is designed to ensure "a continuous source of power indication during the approach to criticality." The design of the ONS wide range instruments is such that they provide indication from 1.0 E-8 to 200% RTP. The monitoring range of the wide range instrument channels, in conjunction with the monitoring range of the source range instrument channels, provides indication throughout the approach to criticality with no reliance upon the power range instruments for this function.

22 The value in NUREG Specifications 3.3.9 and 3.3.10 for THERMAL POWER level as indicated on the wide range neutron flux channels is changed to reflect the appropriate ONS plant specific value.

23 NUREG SR 3.3.1.2 is modified to require a comparison of calorimetric heat balance to power range channel output and adjustment when calorimetric exceeds power range level output by $\geq$ 2%. This is consistent with the description of SR 3.3.1.2 in the NUREG Bases. Current NUREG wording requires a verification that they are within 2% but provides no action if acceptance criteria is not met.

24 The Applicable MODES for Nuclear Overpower High Setpoint function and RCS High Pressure function are expanded to include MODE 3 when not in shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. Note (d) is added to ITS Table 3.3.1-1. This change provides for requirements which are more restrictive than those provided by the CTS. This additional Applicability is appropriate to ensure that the instrumentation required to initiate the insertion of any withdrawn CONTROL RODS is OPERABLE whenever CONTROL RODS are withdrawn or capable of withdrawal. The automatic insertion of any withdrawn CONTROL ROD is consistent with evaluations of accidents initiated from MODE 3. In addition, the applicable MODES for the RCS High Pressure function is modified to apply only in MODE 2 when not in shutdown bypass operation. This is appropriate since the Shutdown Bypass RCS High Pressure function is required to be OPERABLE in MODE 2 during shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. [ONS-020]

25 NUREG 3.3.4, Condition A is modified to specifically state it applies to a diverse trip function being inoperable rather than an undervoltage or shunt trip function being inoperable. This is consistent with the NUREG Bases discussion and the CTS requirements.

26 NUREG Specification 3.3.1 Condition C is revised to specify that these Conditions also apply when two or more required RPS channels are inoperable. This change is made to provide ACTION requirements which specifically remove the unit from the Applicability for this Specification.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if more than one RPS channel is inoperable. Entry into the Required Action of ITS 3.3.1 Condition B, rather than the ACTION requirements of LCO 3.0.3, is more appropriate because specific Required Actions which result in the unit exiting the unique Applicability for each LCO 3.3.1 Function are provided. [ONS-019]

27 NUREG LCO 3.3.14, "EFIC-EFW-Vector Valve Logic," is not adopted since it is not applicable to ONS. ONS design does not include vector valve logic.

28 NUREG Specification 3.3.11, Emergency Feedwater Initiation and Control (EFIC) System Instrumentation; NUREG Specification 3.3.12, EFIC Manual Initiation; and NUREG Specification 3.3.13, EFIC logic, are modified to address Main Steam Line Break Detection and MFW Isolation Circuitry only. ITS Specifications 3.3.14 and 3.3.15 are added to address Emergency Feedwater System Initiation Circuitry and Main Steam Line Break and Main Feedwater Isolation instrumentation separately. The NUREG Specification combines the EFW System Initiation, MSL Isolation and MFW Isolation functions into one Specification apparently due to common instrumentation and similar initiation circuitry. ONS does not have common instrumentation and similar initiation circuitry for these functions. Consistent with CTS, the ITS addresses these requirements by separate Specifications. The Specification titles, LCOs, ACTIONS, and Surveillance Requirements are appropriately modified to reflect ONS specific terminology and design requirements. Where appropriate, ITS Required Actions are based on similar NUREG Required Actions. For example, the Completion Time of one hour for ITS 3.3.15, Required Action A.1 is consistent with NUREG Specification 3.3.7, Required Action A.2, which allows one hour to declare an affected component inoperable when the actuation logic is inoperable.

29 SR 3.3.8.2 is added to the Post Accident Monitoring (PAM) SR Table to capture the 12 month calibration frequency of the containment pressure and hydrogen concentration functions. ITS SR 3.3.8.2 is modified by a note indicating that the SR is only applicable to these two functions. NUREG SR 3.3.17.2 (ONS SR 3.3.8.3) is modified by a note that indicates that the 18 month calibration is not applicable to these two functions. This change is necessary to accommodate the different frequencies for CHANNEL CALIBRATION.

30 NUREG Table 3.3.17-1 (ONS Table 3.3.8-1) is modified to list the Regulatory Guide 1.97 Type A and the Regulatory Guide 1.97 non-Type A instruments and their associated requirements as documented in the NRC

Safety Evaluation Report for Regulatory Guide 1.97 related to Oconee. The "NOTE" at the bottom of the NUREG Table is deleted since it does not apply plant specific.

31 NUREG SR 3.3.11.2, as it relates to the MSLB Detection and MFW Isolation Circuitry, is deleted since the CTS (Table 4.1-1 Item 62) specifies the CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST on the same frequency. Both the CTS and ITS definitions specify that the required calibration includes the CHANNEL FUNCTIONAL TEST. Therefore, the specific requirement to perform the CHANNEL FUNCTIONAL TESTS is not retained in the ITS.

32 The frequency of 92 days for NUREG SR 3.3.15.2 (ITS SR 3.3.16.2) is modified to partially incorporate the CLB. DPC considers the NUREG frequency of 92 days to be inappropriate for ONS. CTS 3.8.10 requires the radiation monitor that initiates purge isolation to be verified operable immediately prior to beginning refueling operations. For consistency with ITS SR 3.9.3.2, which verifies that the reactor building purge supply and exhaust valve actuates to the correct position on an actual or simulated actuation signal once each refueling outage prior to beginning CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the same SR Frequency is adopted for ITS SR 3.3.16.2. This is appropriate since the safety function of the radiation monitor is to isolate the purge valves. Requiring performance of SR 3.3.16.2 at this Frequency represents a reasonable relaxation of the current requirement of "immediately prior to beginning refueling operations."

33 ITS Specifications 3.3.17 through 3.3.22 are added to capture current technical specification requirements for Emergency Power Switching Logic Functions. The EPSL is designed to assure that power is supplied to the unit main feeder buses and, hence to the unit's essential loads. Appropriate LCOs, ACTIONS, and Surveillance Requirements are added.

34 NUREG LCO 3.3.8, "Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)," is not adopted since it is not applicable to ONS. ONS does not use EDGs for emergency power. Comparable ITS requirements related to the Keowee Hydro Units, which are used at Oconee for emergency power, are included in ITS 3.3.19.

35 NUREG Specification 3.3.16, "Control Room Isolation - High Radiation," is not included in the proposed ONS ITS. ONS does not have an automatic Control Room isolation. At ONS, a high radiation alarm is annunciated in the Control Room at which time the Control Room operator can energize the outside air booster fans and filter systems to minimize unfiltered air entering the control room. Adequate administrative controls are in place to ensure the operability of this function.

36 The NUREG applicability for LCO 3.3.15 (ONS ITS LCO 3.3.16) of MODES 1, 2, 3, and 4 is not adopted in the conversion. The ITS requires the

reactor building purge isolation - high radiation monitor to be operable only during CORE ALTERATION and during movement of irradiated fuel assemblies within the containment. At ONS, the reactor building purge valves are required to be verified sealed closed in MODES 1, 2, 3, and 4 and (refer to SR 3.6.3.1). Since the function of the high radiation channel is to initiate closure of these valves on a high radiation signal, the channel need not be OPERABLE during these MODES since the valves are closed.

As a result of the modified applicability, NUREG 3.3.15 ACTIONS A and B are deleted since they no longer apply. These changes maintain the CTS requirements.

37 CTS do not specify an allowable value for the reactor building purge isolation radiation monitor. The UFSAR does not take credit for isolating the purge valves during a refueling accident. The isolation function serves only to minimize radioactive releases but is not required to maintain releases within 10 CFR 100 limits. As such, the specific wording related to the setpoint allowable value in NUREG SR 3.3.15.3 is not included in the ONS ITS.

38 NUREG LCO 3.3.18, "Remote Shutdown System," is not adopted. The ONS CTS does not include any requirements related to shutdown from outside the control room. This function is adequately controlled administratively. In addition, the proposed ONS ITS includes requirements related to the Standby Shutdown Facility, which is designed to mitigate the consequences of postulated fire or flooding incidents, or acts of industrial sabotage to one or more of the three units at Oconee. The SSF is in addition to and supplements the current shutdown capability described in the UFSAR.

39 The allowable value for the RCS Variable Low Pressure Function is located in the Core Operating Limits Report for Oconee. Therefore, the equation with bracketed values provided in NUREG Table 3.3.1-1 for Item 5, RCS Variable Low Pressure, is replaced with: "As specified in the COLR."

40 NUREG SR 3.3.1.11 requires a channel check of the EFW initiation function. A comparable channel check requirement is not included for ITS 3.3.14, EFW Pump Initiation Circuitry. The current test requirements, which do not include a channel check, were adopted and are considered adequate based on operating experience to ensure instrument channel operability.

41 NUREG SR 3.3.1.5 is modified to require the power range channel output to be calibrated to the calorimetric coincident with the imbalance output being calibrated to the imbalance condition predicted by the incore detector system. This is consistent with the requirement actually described in the Bases for SR 3.3.1.5. Since the frequency of this calibration is changed to 31 days, consistent with the CTS

Frequency, the order and numbering of NUREG SRs 3.3.1.4 and 3.3.1.5 are changed to ITS SRs 3.3.1.5 and 3.3.1.4 respectively. A Note is added to NUREG SR 3.3.1.5 similar to those for SR 3.3.1.2 and 3.3.1.3 to allow delay in performance until 24 hours after THERMAL POWER is ≥ 15%. Table 3.3.1-1 is modified to specify additional applicable SRs for Function 1.a & 1.b (SRs 3.3.1.4 and 6), and for Function 8 (SRs 3.3.1.4) and to delete SR 3.3.1.5 as an applicable SR for Function 1.b. A CHANNEL FUNCTIONAL TEST (CFT) of Function 1.b is not currently required and is consider unnecessary since the CHANNEL CALIBRATION includes a CFT and must be performed prior to entry into an applicable MODE or condition in which Function 1.b is required to be OPERABLE.

NUREG SR 3.3.1.5 is called a CHANNEL CALIBRATION but is not actually a calibration of the Function, only a calibration of the inputs to the function (i.e., calibrates the power range channels to the incore channels). SR 3.3.1.5 is re-worded to specifically require the power range channels be calibrated rather than to incorrectly specify a CHANNEL CALIBRATION. As such, there are no requirements to demonstrate OPERABILITY of the entire instrument channel. Therefore, requirements are added to perform a CHANNEL CALIBRATION every 18 months for Functions 1.a & 1.b and to perform a CHANNEL FUNCTIONAL TEST every 45 days on a STAGGERED TEST BASIS for Functions 1.a and 8. The note to NUREG SR 3.3.1.5 to allow delay in performance until 24 hours after THERMAL POWER is ≥ 15% RTP is necessary since at lower power levels calorimetric data are inaccurate and the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. NUREG SR 3.3.1.5 applicability to Function 1.b is deleted since the Nuclear Overpower Low Setpoint is only applicable at power levels < 5% RTP. [ONS-001]

42 Specification 3.3.3 Required Action B.2.2 & C.2 and Specification 3.3.4 Required Action D.2.2 and E.2 are modified to specify: "Remove power from all CRD trip breakers" from "Remove all power to CRD System." Corresponding Bases revised accordingly. The requirement to remove all power to the CRD system could be interpreted to include all control power and logic cabinets since they are a part of this system. It is more appropriate to remove power from all CRD trip breakers. This action places the unit in condition where the LCO no longer applies. [ONS-002]

43 The Note modifying NUREG-1430 SR 3.3.3.1 is deleted. This Note allows a delay of up to 8 hours for the entry into the Conditions and Required Actions for the performance of this surveillance. At ONS, the reactor trip module (RTM) cannot be bypassed. Performance of this SR, does not render the RTM inoperable. Therefore, the NUREG SR Note is inappropriate. During performance of the referenced CHANNEL FUNCTIONAL TEST, the RTM is either in service or is in the tripped condition. In either case, the RTM is capable of performing its design functions of receiving trip status from and sending trip status to the other RTMs and is capable of tripping its associated trip device(s). [ONS-003]

44 CTS Table 4.1-1, Item 1 specifies a monthly functional test for the Reactor Trip Modules. In the conversion to ITS, this Frequency is retained in ITS SR 3.3.3.1. This change is made to provide requirements consistent with CTS for this testing. No new requirements are added by this change and no existing requirements are removed.

45 NUREG Specification 3.3.17 is modified to incorporate plant specific requirements for HPI, LPI and RBS flow instrument channels and BWST water level instrument channels. At ONS there is only one flow instrument channel per train. If the flow instrument is inoperable, ONS considers the associated train inoperable since without flow indication the operator has no means of precluding pump runout or loss of NPSH. Therefore, the appropriate action for an inoperable flow channel is to declare the affected train inoperable. ACTION F is added to address the condition where one or more of these required flow instrument channels is inoperable. Required Action F.1 requires the affected train be declared inoperable and the appropriate action entered for the affected system. Condition F has a note that indicates that the Condition only applies to the flow instrument channels (Function 18, 19, and 20). The addition of ACTION F made it necessary to modify Conditions A, C, and D to exclude or include Functions for which the conditions are applicable. Table 3.3.17-1 is modified to replace the Condition reference from the Required Action E.1 as being not applicable since the appropriate action only applies to these particular instruments and that action is contained within Required Action F.1.

ACTION E was added to address the condition where one of two BWST water level channels is inoperable. Continuous operation with one of the two required channels is not appropriate because alternate indications are not available. This indication is crucial in determining when the water source for ECCS should be swapped from the BWST to the reactor building sump. Therefore, 24 hours is allowed to restore the indication, consistent with CTS requirements. With both BWST water level channels inoperable, the appropriate action is to shut down.

46 NUREG 3.3.1 Required Action E.1, G.1 and Table 3.3.1-1 provide a bracketed value for the Applicability of the Main Turbine Trip and Loss of Main Feedwater Pumps RPS Functions. The ITS Applicability for each function is based on analyses presented in BAW 1893 that show for operation below certain power levels, the trips are not necessary to minimize challenges to the PORV as required by NUREG-0737. The CTS Applicability for these functions is when the reactor is in a startup mode or in a critical state. Duke Energy has performed a plant specific analysis which concludes that the Oconee RPS System is consistent with the BAW analyses and that the appropriate plant specific applicability for each function is 30% RTP for the Main Turbine Trip function and 2% RTP for the Loss of Main Feedwater Pumps trip function.

47 NUREG Table 3.3.1-1 provides a bracketed value for allowable value for the Main Turbine Trip (Hydraulic Fluid Pressure) function and the Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) function. CTS does not specify allowable values for these functions. Appropriate plant specific values are added.

48 ITS Specification 3.3.22 is added to require the Manual Keowee Emergency Start Function to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies. This addition is necessitated by the addition of requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies (refer to Section 3.8). Required Action A.1 requires both Keowee Hydro Units to be declared inoperable immediately when the required channel is inoperable. ITS SR 3.3.22.1 requires a CHANNEL FUNCTIONAL TEST of the Keowee manual emergency start function every 12 months. The EPSL is designed to assure that power is supplied to the unit main feeder buses and, hence to the unit's essential loads.

49 ONS design requires that the voltage sensing circuit associated with an AC power source be OPERABLE for the AC power source to be considered OPERABLE. Therefore, since requirements for AC Source in MODES 5 and 6 and during movement of irradiated fuel assemblies are added (refer to Section 3.8), requirements for EPSL voltage sensing circuits are included in the ITS. LCO Note 2 is added to specify that only the EPSL voltage sensing circuit(s) associated with required AC power source(s) are required to be OPERABLE. ITS 3.3.18 ACTION C is added to require the affected AC Source to be declared inoperable when the Required Action and associated Completion Time is not met or when two or more channels of a required circuit(s) are inoperable in MODES 5 and 6. ITS 3.3.18 ACTION D is added to require suspending movement of irradiated fuel assemblies when the Required Action and associated Completion Time is not met during movement of irradiated fuel assemblies.

50 NUREG Specification 3.3.11 Condition F is revised to specify that this Condition also applies when two or more channels are inoperable. NUREG 3.3.12 Condition C is revised to specify that this Condition also applies when two manual initiation switches are inoperable. NUREG 3.3.13 Condition B is revised to specify that this Condition also applies when two logic channels are inoperable. These changes are made to provide ACTION requirements which specifically remove the unit from the Applicability for these Specification.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if more than one channel (instrument or logic) or manual initiation switch is inoperable. Entry into the Required Action of the Specification, rather than the ACTION requirements of LCO 3.0.3, is more appropriate because specific Required Actions which result in the unit exiting the Applicability for the LCO are provided. [ONS-019]

OCONEE NUCLEAR STATION

IMPROVED TECHNICAL SPECIFICATION CONVERSION

SECTION 3.3 - INSTRUMENTATION

ATTACHMENT 6

NUREG 1430 MARKUP AND JUSTIFICATIONS

BASES

B 3.3 INSTRUMENTATION (4)

B 3.3.1 Reactor Protect~~ion~~ve System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip to protect against violating the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during anticipated ~~operational occurrences (AOOs)~~ transients (4). By tripping the reactor, the RPS also assists the Engineered ~~Safety Feature~~ Safeguards (ES~~F~~) (4) Systems in mitigating accidents.

The protecti~~on~~ve (4) and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by specifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the LCOs on other reactor system parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs, establishes the threshold for protective system action to prevent exceeding acceptable limits during ~~Design Basis Accidents (DBAs)~~ accidents or transients (4).

During ~~AOOs~~ anticipated transients (4), which are those events expected to occur one or more times during the unit's life, the acceptable limit is:

a. The departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value;

b. Fuel centerline melt shall not occur; and

c. The RCS pressure SL of 2750 psia shall not be exceeded.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 20 and 10 CFR 100 criteria during ~~AOOs~~ anticipated transients.

Accidents are events that are analyzed even though they are not expected to occur during the unit's life. The acceptable limit during accidents is that the offsite dose shall be maintained within 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

reference (19)

(continued)

BASES

BACKGROUND (continued)

RPS Overview

The RPS consists of four separate redundant protection ve 4 channels that receive inputs of neutron flux, RCS pressure, RCS flow, RCS temperature, RCS pump status, reactor building (RB) pressure, main feedwater (MFW) pump status, and turbine status.

Figure ~~[ ]~~ 7.1, ~~FSAR~~ 2, Chapter ~~[7]~~ (Ref. 1), ve 4 shows the arrangement of a typical RPS protection channel. A protection ve channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and ~~CONTROL~~ ROD drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection ve System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection ve System (RPS) – Reactor Trip Module (RTM)," and LCO 3.3.4, "CONTROL ROD Drive (CRD) Trip Devices," discuss the remaining RPS elements. 1 4 4

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of any two trip signals in any of the four RPS channels will result in the trip of the reactor.

The Reactor Trip System (RTS) contains multiple CRD trip devices; two AC trip breakers, ~~and~~ two DC trip breaker pairs, ~~that provide a path for power to the CRD System. Additionally, the power for most of the CRDs passes through electronic trip assembly (ETA) relays.~~ and eight electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having ~~either two breakers or a breaker and an ETA relay in series~~ one AC breaker in series with a pair of DC breakers and functionally in series with four ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs. Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip. 4

The RPS consists of four independent protection ve 4 channels, each containing an RTM. The RTM receives signals from its own measurement channels that indicate a protection ve channel 4 trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic

(continued)

BASES

BACKGROUND

RPS Overview (continued)

of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip ~~breaker~~ device (2) and ETA relays (2).

The reactor is tripped by opening circuit breakers that interrupt the control (2) power supply to the CRDs. Six breakers are installed to increase reliability and allow testing of the trip system. A one-out-of-two taken twice logic is used to interrupt power to the rods.

The RPS has ~~two~~ three bypasses: a shutdown bypass, a dummy bistable and ~~a~~ channel bypass. Shutdown bypass allows the withdrawal of ~~safety~~ GnRPS rods for SDM availability and rapid negative reactivity insertion during unit cooldowns or heatups. ~~A~~ The RPS Channel bypass ~~is used~~ allows one entire RPS channel to be taken out of service for maintenance and testing. ~~Test circuits in the trip strings allow complete testing of all RPS trip Functions.~~ The dummy bistable is used to bypass one or more functions (bistable trips) associated with one RPS channel.

The RPS operates from the instrumentation channels discussed next. The specific relationship between measurement channels and protecti~~on~~ve channels differs from parameter to parameter. Three basic configurations are used:

a. Four completely redundant measurements (e.g., reactor coolant flow) with one channel input to each protecti~~on~~ve channel;

b. Four channels that provide similar, but not identical, measurements (e.g., power range nuclear instrumentation where each RPS channel monitors a different quadrant), with one channel input to each protecti~~on~~ve channel; and

c. Redundant measurements with combinational trip logic outside of the protecti~~on~~ve channels and the combined output provided to each protecti~~on~~ve channel (e.g., main ~~turbine~~ feedwater pump trip instrumentation).

These arrangements and the relationship of instrumentation channels to trip Functions are discussed next to assist in understanding the overall effect of instrumentation channel failure.

(continued)

BASES

BACKGROUND (continued)

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following trip Functions:

1. Nuclear Overpower
   a. Nuclear Overpower – High Setpoint;
   b. Nuclear Overpower – Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower Flux/Flow Imbalance ~~RCS Flow and Measured AXIAL POWER IMBALANCE (Power Imbalance Flow)~~; (4)

9. Main Turbine Trip (~~Control Oil~~ Hydraulic Fluid Pressure)~~; and~~

10. Loss of Main Feedwater (LOMFW) Pumps (~~Control~~ Hydraulic Oil Pressure).

The power range instrumentation has four linear level channels, one for each core quadrant. Each channel feeds one RPS protective ~~protection~~ channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. (4) The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The difference of the top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE for the associated core quadrant ~~of the reactor core~~. (2)

Reactor Coolant System Outlet Temperature

The Reactor Coolant System Outlet Temperature provides input to the following Functions:

2. RCS High Outlet Temperature; and

5. RCS Variable Low Pressure.

The RCS Outlet Temperature is measured by two resistance elements in each hot leg, for a total of four. One temperature detector is associated with each protective ~~protection~~ channel. (4)

(continued)

BASES

BACKGROUND (continued)

Reactor Coolant System Pressure

The Reactor Coolant System Pressure provides input to the following Functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

The RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each hot leg, for a total of four. One sensor is associated with each protect~~ion~~ve channel. (4)

Reactor Building Pressure

The Reactor Building Pressure measurements provide input only to the Reactor Building High Pressure trip, Function 6. There are four RB High Pressure sensors, one associated with each protect~~ion~~ve channel. (4)

Reactor Coolant Pump Power Monitoring

Reactor coolant pump power monitors are inputs to the Reactor Coolant Pump to Power trip, Function 7. Each RCP, operating current, and voltage is measured by four current transformers and four potential transformers driving ~~four overpower and~~ four underpower relays. Each power monitoring channel consists of ~~an overpower relay and~~ an underpower relay. One channel for each pump is associated with each protect~~ion~~ve channel. (4)

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower Flux/Flow Imbalance ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ (4) trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight high accuracy differential pressure transmitters, four on each loop, which measure flow

(continued)

BASES

BACKGROUND

Reactor Coolant System Flow (continued)

through calibrated flow tubes. One flow input in each loop is associated with each protection ve channel. 4

Main Turbine Automatic Stop Oil Pressure Hydraulic Fluid

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip (~~Control Oil~~ Pressure) reactor trip, Function 9. Each of the four protection ve channels receives turbine status information from the ~~same~~ four pressure switches monitoring main turbine automatic stop oil pressure. one of 2 An open indication will be provided to the RPS on a turbine trip. Contact buffers in each protection ve channel 4 continuously monitor the status of the contact inputs and initiate an RPS trip when a main turbine trip is indicated. 2

Feedwater Pump ~~Control~~ Hydraulic Oil Pressure 4

Feedwater Pump ~~Control~~ Oil Pressure is an input to the Loss of Main Feedwater Pumps (~~Control~~ Oil Pressure) trip, Function 10. ~~Control~~ oil pressure is measured by four switches on each feedwater pump. One switch on each pump is connected in series with a switch on the other MFW pump, 4 associated with each protection ve channel. 4

RPS Bypasses

The RPS is designed with ~~two~~ three types of bypasses: dummy bistables, 4 channel bypass and shutdown bypass.

4 INSERT B3.3-6A

Channel bypass provides a method of placing all Functions in one RPS protection ve channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed next. 4

4 INSERT B3.3-6B

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protection ve channel trip relay energized regardless of the status of the instrumentation channel of 4

(continued)

INSERT B 3.3-6A

The dummy bistable provides a method of placing one or more functions in a RPS protective channel in a bypassed condition,

INSERT B 3.3-6B

Dummy Bistable

The dummy bistable is used to bypass one or more functions (bistable trips) associated with one RPS Channel. A dummy bistable is used if a parameter in an RPS channel fails and causes that channel to trip. Dummy bistables may be used in only one RPS channel at a time. Also, if an RPS channel is bypassed, no other RPS channel may contain a dummy bistable. Inserting a dummy bistable in the place of a failed (tripped) bistable allows the RPS channels to be reset, thus allowing the remainder of the functions in that RPS channel to be returned to service. This is more conservative than manually bypassing the entire RPS channel. The trip functions in an RPS channel with a dummy bistable are not considered OPERABLE.

2 or otherwise inoperable

2 can be verified by observing alarms/indicator lights.

BASES

BACKGROUND

Channel Bypass (continued)

5

the bistable relay contacts. To place a protection channel in channel bypass, the other ~~three~~ channels must not be in channel bypass. This ~~is ensured by contacts from the other channels being in series with the channel bypass relay~~. ~~If any contact is open, the second channel cannot be bypassed. The second condition is the closing of the key switch. When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition.~~ All RPS trips are reduced to a two-out-of-three logic in channel bypass. 4 ve 2

This is administratively controlled by having only one manual bypass key available for each unit. 2

Shutdown Bypass 2 and heatup

During unit cooldown, it is desirable to leave the safety rods withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

4 at least partially

However, the unit is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at 1800 psig and the rods will fall into the core. To avoid this, the protection system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1800 psig. The RCS pressure is decreased to less than 1720 psig, then each RPS channel is placed in shutdown bypass.

2 when

In shutdown bypass, a normally closed contact opens ~~and~~ the operator closes the shutdown bypass key switch. This action bypasses the RCS Low Pressure trip, Nuclear Overpower ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ Flux/Flow Imbalance trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, and inserts a new RCS High Pressure, ~~1850~~ psig trip. The operator can now withdraw the safety rods for additional ~~SDM~~. 4

4 1720

rapidly insertable negative reactivity 2

The insertion of the new high pressure trip performs two functions. First, with a trip setpoint of 1720 psig, the bistable prevents operation at normal system pressure, 2155 psig, with a portion of the RPS bypassed. The second

(continued)

BASES

BACKGROUND

Shutdown Bypass (continued)

function is to ensure that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. The safety rods are then withdrawn and remain at the full out condition for the rest of the heatup.

In addition to the Shutdown Bypass RCS High Pressure trip, the high flux trip setpoint is administratively reduced to (2) (≤) 5% RTP prior to placing ~~while~~ the RPS ~~is~~ in shutdown bypass. This provides a backup to the Shutdown Bypass RCS High Pressure trip and allows low ~~temperature~~ power physics testing while preventing the generation of any significant amount of power. (2)

Module Interlock and Test Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode, it causes the test trip relay to open and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested.

(4) INSERT B3.3-8A

Trip Setpoints/Allowable Value

The trip setpoints are the normal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy. (i.e., ~~±~~ [rack calibration ~~+~~ comparator setting accuracy]) (2)

The trip setpoints used in the bistables are based on the analytical limits stated in UFSAR, Chapter 15 ~~[14]~~ (Ref. 2). (1) The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 3), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively

(continued)

INSERT B 3.3-8A

Each trip module is electrically interlocked to the other three trip modules. Removal of a trip module will indicate a tripped channel in the remaining trip modules.

BASES

BACKGROUND

Trip Setpoints/Allowable Value (continued)

adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in ~~"[Unit Specific Setpoint Methodology]" (Ref. 4)~~ Reference 4. ~~The actual nominal trip setpoint entered into the bistable~~ is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during ~~AOOs~~ anticipated transients and that the consequences of ~~DBAs~~ accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the ~~AOO~~ anticipated transient or ~~DBA~~ accident and the equipment functions as designed. Note that in LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 for Functions 1 through 8 and 11 are the LSSS.

Each channel can be tested online to verify that the ~~signal and~~ setpoint accuracy ~~are~~ is within the specified allowance requirements ~~of Reference 4~~. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. ~~The process equipment for the channel in test is then tested, verified, and calibrated.~~ Surveillances for the channels are specified in the SR section.

The Allowable Values listed in Table 3.3.1-1 are based on the methodology described in ~~"[Unit Specific Setpoint Methodology]" (Ref. 4)~~ Reference 4, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of those uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

(continued)

UFSAR Chapter 15 (2)

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in (Ref. ~~[2]~~ 1) takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high RCS temperature, turbine trip, and loss of main feedwater. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions also serve as backups to Functions that were credited in the safety analysis. (1)

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. ~~The four~~ three channels of each Function in Table 3.3.1-1 of the RPS instrumentation shall be OPERABLE ~~at all times the reactor is critical~~ during its Specified Applicability to ensure that a reactor trip will be actuated if needed. Additionally, during shutdown bypass with any CRD trip breaker closed, the applicable RPS Functions must also be available. This ensures the capability to trip the withdrawn CONTROL RODS exists at all times that rod motion is possible. The trip Function channels specified in Table 3.3.1-1 are considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1. (5) (2)

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel bypass. Bypass effectively places the unit in a two-out-of-three logic configuration that can still initiate a reactor trip, even with a single failure within the system. (5)

Only the Allowable Values are specified for each RPS trip Function in the LCO. Nominal trip setpoints are specified in the ~~unit specific~~ setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable (2)

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

provided that operation and testing are consistent with the assumptions of the ~~unit specific~~ setpoint calculations. Each Allowable Value specified is more conservative than instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "[Unit Specific Setpoint Methodology]" (Ref. 4). Reference 4 2

For most RPS Functions, the trip setpoint Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific ~~figures~~ values for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the minimum deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

   a. Nuclear Overpower - High Setpoint

      The Nuclear Overpower - High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core fast neutron leakage flux. 2

      The Nuclear Overpower - High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to a maximum value of the design overpower. prevent exceeding acceptable fuel damage limits 4

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Nuclear Overpower - High Setpoint (continued)

4

Thus, the Nuclear Overpower - High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower Flux/Flow Imbalance ~~RCS Flow and Measured AXIAL POWER IMBALANCE,~~ provide more direct protection. The role of the Nuclear Overpower - High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower - High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident, and the rod ejection accident ~~and the steam line break accident~~. By providing a trip during these events, the Nuclear Overpower - High Setpoint trip protects the unit from excessive power levels and also serves to ~~reduce~~ limit reactor power to prevent violation of the RCS pressure SL. 4

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

~~The specified Allowable Value is selected to ensure that a trip occurs before reactor power exceeds the highest point at which the RCS Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trips are analyzed to provide protection against DNB and fuel centerline melt. The Allowable Value does not account for harsh environment induced errors, because the trip will actuate prior to degraded environmental conditions being reached.~~

26

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Nuclear Overpower - Low Setpoint

Prior to initiating ~~While in~~ shutdown bypass~~, with the Shutdown Bypass RCS High Pressure trip OPERABLE, the~~ (4) Nuclear Overpower - Low Setpoint trip must be reduced to ≤ 5% RTP. The low power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the unit is protected from excessive power conditions when other RPS trips are bypassed.

The setpoint Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. RCS High Outlet Temperature

The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for DNB. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable Low Pressure trip. The RCS High Outlet Temperature trip provides steady state protection for the DNBR SL.

The RCS High Outlet Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip is not required to mitigate accidents that create harsh conditions in the RB.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. RCS High Pressure

relief 4

The RCS High Pressure trip works in conjunction with the pressurizer and main steam ~~safety~~ valves to prevent RCS overpressurization, thereby protecting the RCS High Pressure SL.

The RCS High Pressure trip has been credited in the ~~accident~~ analysis calculations for slow positive reactivity insertion transients (rod withdrawal ~~accidents~~ and moderator dilution) ~~and loss of feedwater accidents.~~ The rod withdrawal ~~accidents~~ cover a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower – High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

4 transient

transients

4

transient

4

The setpoint Allowable Value is selected to ensure that the RCS High Pressure SL is not challenged during steady state operation or slow power increasing transients. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions because the equipment is not required to mitigate accidents that create harsh conditions in the RB.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the RCS High Outlet Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system pressure approaches the conditions necessary for DNB. The RCS Low Pressure trip provides DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure setpoint Allowable Value is selected to ensure that a reactor trip occurs before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. RCS Low Pressure (continued)

credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs) and main steam line break (MSLB) accidents. ~~Consequently,~~ Harsh RB conditions created by small break LOCAs ~~can~~ not affect performance of the RCS pressure sensors and transmitters within the time frame for a reactor trip. Therefore, degraded environmental conditions are not considered in the Allowable Value determination. 4

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the RCS High Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the system parameters of pressure and temperature approach the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the RCS High Outlet Temperature within the range specified by the RCS High Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure setpoint Allowable Value is selected to ensure that a trip occurs when temperature and pressure approach the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is ~~not~~ assumed for transient protection in the unit safety analysis but does not affect the limiting cases; therefore, determination of the setpoint Allowable Value does not account for errors induced by a harsh RB environment. 4

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. It also provides a backup for RPS trip instruments exposed to an RB HELB environment.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Reactor Building High Pressure (continued)

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients inside containment. ~~The components are exposed to high radiation conditions.~~ Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.

4

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing are insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline melt SLs.

4

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of ~~four~~ more than two RCPs. The trip also provides the primary ~~protection for the~~ loss of a pump or pumps, which would result in both pumps in a single steam generator loop being tripped.

4

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least three RCPs are operating. RCP status is monitored by power transducers on each pump. These relays indicate a loss of an RCP on overpower with an Allowable Value of ~~≥ 14,400 kW~~ and on underpower with an Allowable Value of ≤ 1752 kW. The overpower Allowable Value is selected low enough to detect locked rotor conditions

14

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Reactor Coolant Pump to Power (continued)

(although credit is not allowed for this capability) but high enough to avoid a spurious trip on the in rush current when the pumps start. 14 ~~The underpower Allowable Value~~ is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power ~~Allowable Value~~ setpoint account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment. 4

8. Nuclear Overpower Flux/Flow Imbalance ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~

The Nuclear Overpower Flux/Flow Imbalance ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ trip provides steady state protection for the power imbalance SLs. A reactor trip is initiated ~~when~~ prior to the core power, AXIAL POWER IMBALANCE, ~~and~~ reactor coolant flow conditions ~~indicate an approach to~~ exceeding the DNB or fuel centerline ~~melt~~ temperature limits. 2

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the DNBR SL for the loss of ~~a single~~ one or more RCP and for locked RCP rotor accidents. ~~The imbalance portion of the trip is credited for steady state protection only.~~ 4

The power to flow ratio of the Nuclear Overpower Flux/Flow Imbalance ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system flow rate is less than full four pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs when the core power, axial power peaking, and

(continued)

Flux/Flow Imbalance 4

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Nuclear Overpower ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ (continued)

temperature 2

reactor coolant flow conditions indicate an approach to DNB or fuel centerline ~~melt~~ limits. By measuring reactor coolant flow and by tripping only when conditions approach an SL, the unit can operate with the loss of one pump from a four pump initial condition. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

4 at power levels at least as low as approximately 80% RTP

Hydraulic Fluid 4

9. Main Turbine Trip (~~Control Oil~~ Pressure)

The Main Turbine Trip Function trips the reactor when the main turbine is lost at high power levels. The Main Turbine Trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine Trip Function was added to the B&W designed units in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS power operated relief valve (PORV) actuation for turbine trip cases. This trip is activated at higher power levels, thereby limiting the range through which the Integrated Control System must provide an automatic runback on a turbine trip.

hydraulic fluid 4

Each of the four turbine ~~oil~~ pressure switches feeds one ~~all four~~ protection channels through buffers that continuously monitor the status of the contacts. Therefore, failure of any pressure switch affects all protection channels. 4

Hydraulic Fluid

For the Main Turbine Trip (~~Control Oil~~ Pressure) bistable, the Allowable Value of ~~45~~ psig is selected to provide a trip whenever ~~feedwater pump~~ ~~control oil~~ pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of ~~45%~~ RTP. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

5 800

2 main turbine

4 hydraulic fluid

5 30%

(continued)

BASES

4

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Loss of Main Feedwater Pumps (~~Control~~ Hydraulic Oil Pressure)

The Loss of Main Feedwater Pumps (~~Control~~ Oil Pressure) trip provides a reactor trip at high power levels when both MFW pumps are lost. The trip provides an early reactor trip in anticipation ~~of the~~ 2 loss of heat sink associated with the LOMFW. This trip was added in accordance with NUREG-0737 (Ref. 5) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip at high power levels for a LOMFW to minimize challenges to the PORV.

2 hydraulic 4 75 5

For the feedwater pump ~~control~~ oil pressure bistables 2 the Allowable Value of ~~55~~ psig is selected to provide a trip whenever feedwater pump ~~control~~ oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 4 5 - 2 ~~15%~~ RTP. The Loss of Main Feedwater Pumps (~~Control~~ Hydraulic Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass RCS High Pressure is provided to allow for withdrawing the CONTROL RODS prior to reaching the normal RCS Low Pressure trip setpoint. The shutdown bypass provides trip protection during deboration and RCS heatup by allowing the operator to 4 at least partially withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Use of the shutdown bypass trip requires that the neutron power trip setpoint be reduced to 5% of full power or less. The Shutdown Bypass RCS High Pressure trip forces a reactor trip to occur whenever the unit switches from power operation to shutdown bypass or vice versa. This ensures that the CONTROL RODS are all inserted 2 ~~and the flux distribution is known~~ before power operation can begin. The operator is required to remove the shutdown bypass, reset the Nuclear

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

Overpower – High Power trip setpoint, and again withdraw the safety ~~rod~~ groups before proceeding with startup. 2 u

Accidents analyzed ~~in~~ the ~~F~~SAR, Chapter ~~[14]~~ 15 (Ref. 2), 1 do not describe events that occur during shutdown bypass operation, because the consequences of these events are enveloped by the events presented in the u ~~F~~SAR. 2

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ ~~[~~1720~~]~~ psig and the Nuclear Overpower – Low Setpoint set at or below 5% RTP, the trips listed below can be bypassed. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower – Low Setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary. 1

1.a Nuclear Overpower – High Setpoint; 3. RCS High Pressure; 4

4. RCS Low Pressure;

5. RCS Variable Low Pressure; 4

7. Reactor Coolant Pump to Power; and Flux/Flow Imbalance

8. Nuclear Overpower ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~.

The Shutdown Bypass RCS High Pressure Function's Allowable Value is selected to ensure a trip occurs before producing THERMAL POWER.

2 General Discussion

The RPS satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 8). 2

In MODES 1 and 2, the following trips shall can be 2 be OPERABLE because the reactor ~~is~~ can be critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during ~~AOOs~~ anticipated transients 4 and to assist the ~~ESFAS~~ ESPS 4 in providing acceptable consequences during accidents.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued) 2

1.a Nuclear Overpower - High Setpoint;

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower Flux/Flow Imbalance ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~. 4 5

5 INSERT B 3.3-21A

Functions 1, 3, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below ~~[1720]~~ psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower - Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower - Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary. 1

~~Two other Functions are required to be OPERABLE during portions of MODE 1. These are the Main Turbine Trip (Control Oil Pressure) and the Loss of Main Feedwater Pumps (Control Oil Pressure) trip. These Functions are required to be OPERABLE above [45]% RTP and [15]% RTP, respectively. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).~~ 5

Because the ~~only~~ safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5 if either the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 ~~when~~ because the CONTROL RODS are normally decoupled from the CRDs. 2

(continued)

INSERT B 3.3-21A

In MODE 3 when not operating in shutdown bypass but with any CRD trip breaker in the closed position and the CRD system capable of rod withdrawal, the Nuclear Overpower–High Setpoint trip and the RCS High Pressure trip are required to be OPERABLE.

The Main Turbine Trip (Hydraulic Fluid Pressure) Function is required to be OPERABLE in MODE 1 at ≥ 30% RTP. The Loss of Main Feedwater Pumps (Hydraulic Oil Pressure) Function is required to be OPERABLE in MODE 1 and in MODE 2 at ≥ 2% RTP. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)	However, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips are sufficient to prevent an approach to conditions that could challenge SLs.

ACTIONS

Conditions A, and B ~~, and C~~ are applicable to all RPS protective ~~on~~ Functions. ~~If a channel~~'s trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and Condition A ~~or Conditions A and B~~ entered immediately. 5

5 ~~When the number of inoperable channels in a trip Function exceed those specified in the related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.~~

13 ~~[Reviewer's Note: If a unit is to take credit for topical reports as the basis for justifying Completion Times, the reports must be supported by an NRC Staff Safety Evaluation Report (SER) that establishes the acceptability of each topical report for that unit.]~~

5 ~~A.1~~

~~If one or more Functions in one protection channel become inoperable, the affected protection channel must be placed in bypass or trip. If the channel is bypassed, all RPS Functions are placed in a two-out-of-three logic configuration and the bypass of any other channel is prevented. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS Functions in a one-out-of-three configuration.~~

(continued)

(5) <Entire Page>

BASES

ACTIONS

A.1 (continued)

Operation in the two-out-of-three configuration or in the one-out-of-three configuration may continue indefinitely based on the NRC SER for BAW-10167, Supplement 2 (Ref. 7). In this configuration, the RPS is capable of performing its trip Function in the presence of any single random failure. The 1 hour Completion Time is sufficient to perform Required Action A.1.

A ~~B~~.1 ~~and B.2~~

For Required Action A ~~B~~.1 ~~and Required Action B.2~~, if one or more Functions ~~in two protection~~ in a required protective channel~~s~~ become~~s~~ inoperable, ~~one of two inoperable protection~~ the affected protective channel~~s~~ must be placed in trip ~~and the other in bypass~~. Th~~ese~~is Required Action~~s~~ places all RPS Functions in a one-out-of-two logic configuration ~~and prevent bypass of a second channel~~. In this configuration, the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1 hour Completion Time is sufficient time to perform Required Action A ~~B~~.1 ~~and Required Action B.2~~.

(20) INSERT B3.3-23A

B ~~C~~.1

Required Action B ~~C~~.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. ~~Each time an inoperable channel has not met any~~ If the Required Action~~s~~ of Condition A ~~or B, as applicable, and the~~ and associated Completion Time ~~has expired~~ are not met or if more than two channels are inoperable, Condition B ~~C~~ is ~~entered for that channel and~~ provide~~s~~ to for transfer to the appropriate subsequent Condition.

C ~~D~~.1 and C ~~D~~.2

If the Required Action and associated Completion Time of Condition A ~~or B~~ are not met and Table 3.3.1-1 directs entry into Condition C ~~D~~, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of ~~6~~ hours is reasonable, based on operating experience, to reach MODE 3 (12)

(continued)

INSERT B 3.3-23A

The "non-required" channel is placed in bypass when the required inoperable channel is placed in trip to prevent bypass of a second required channel.

BASES

ACTIONS

~~B~~ C.1 and ~~B~~ C.2 (continued)

from full power conditions in an orderly manner and to open all CRD trip breakers without challenging ~~plant~~ unit systems. (2)

~~E~~ D.1

If the Required Action and associated Completion Time of Condition A ~~or B~~ are not met and Table 3.3.1-1 directs entry into Condition ~~E~~ D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging ~~plant~~ unit systems. (2)

~~F~~ E.1

If the Required Action and associated Completion Time of Condition A ~~or B~~ are not met and Table 3.3.1-1 directs entry into Condition ~~F~~ E, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < ~~[45]~~ 30% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach ~~[45]~~ 30% RTP from full power conditions in an orderly manner without challenging ~~plant~~ unit systems. (2)

~~G~~ F.1

If the Required Action and associated Completion Time of Condition A ~~or B~~ are not met and Table 3.3.1-1 directs entry into Condition ~~G~~ F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced < ~~[15]~~ 2% RTP. The allowed Completion Time of ~~6~~ 12 hours is reasonable, based on operating experience, to reach ~~[15]~~ 2% RTP from full power conditions in an orderly manner without challenging ~~plant~~ unit systems. (5) (2)

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION ~~and RPS RESPONSE TIME~~ testing. 5

The SRs are modified by a Note. The ~~first~~ Note directs 2 the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

13

~~Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequencies are based on approved topical reports. For a licensee to use these times, the licensee must justify the Frequencies as required by the NRC Staff SER for the topical report.~~

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

2

Agreement criteria are determined ~~by the unit staff~~ based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.

equivalent to — 2

The Frequency, ~~about once~~ every shift, is based on operating experience that demonstrates channel failure is rare. Since

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower ~~RCS Flow and Measured AXIAL POWER IMBALANCE~~ Flux/Flow Imbalance Function, the CHANNEL CHECK must be performed on each input. 4

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 24 hours when reactor power is > 15% RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range channels are normalized to the calorimetric. If the calorimetric exceeds the Nuclear Instrumentation System (NIS) channel output by ≥ [2]% RTP, the NIS is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. A Note clarifies that this Surveillance is required ~~only if~~ to be performed reactor power is ≥ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are ~~inaccurate~~ less accurate. 1 2

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric exceeds the power range channel's output by ≥ [2]% RTP. The value of [2]% is adequate because this value is assumed in the safety analyses of USAR, Chapter ~~[14]~~ 15 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 24 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds 1 2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued)

a small fraction of {2}% in any 24 hour period. (1)
Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is > 15% RTP. A Note clarifies that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. If the absolute difference between the power range and incore measurements is ≥ {2}% RTP, the power range channel is not inoperable, (1) but a ~~CHANNEL~~ CALIBRATION that adjusts the measured imbalance to agree with the incore measurements is (27) necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SR 3.3.1.~~4~~ 5 <Change order>

A CHANNEL FUNCTIONAL TEST is performed on each required RPS channel to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current ~~unit specific~~ setpoint analysis.

(2) The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.~~4~~5 (continued)

surveillance interval extension analysis. The requirements for this review are outlined in BAW-10167 (Ref. ~~8~~ 7). 2

The Frequency of ~~[~~45~~]~~ days on a STAGGERED TEST BASIS is consistent with the calculations of Reference 7 that indicate the RPS retains a high level of reliability for this test interval. 1

27

SR 3.3.1.~~5~~4 <Change order>

calibrates

calibrated

This SR ~~is the performance of a CHANNEL CALIBRATION every [92] days. This CHANNEL CALIBRATION normalizes~~ the power range channel output to the calorimetric coincident with the imbalance output being ~~normalized~~ to the imbalance condition ~~predicted~~ determined by the incore neutron detector system.

27

~~The calibration for both imbalance and total power is integrated in the power imbalance detector calibration procedure.~~ The ~~[92]~~ day Frequency specified for the Nuclear Overpower trip string ~~is consistent with the drift assumptions made in the "[Unit Specific Setpoint Methodology]" (Ref. 4)~~. Furthermore, operating experience shows the reliability of the trip string is acceptable when calibrated on this interval. ~~A Note clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.~~

27 INSERT B 3.3-28A

31 1

considers drift determined using the methodology of Reference 4

2

27

SR 3.3.1.6

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

(continued)

INSERT B 3.3-28A

A Note clarifies that this Surveillance is required to be performed only if reactor power is $\geq$ 15% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are less accurate.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.6 (continued)

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the ~~unit specific~~ setpoint analysis. 2

TSTF-019, R1

INSERT B3.3-29A

The Frequency is justified by the assumption of an ~~[18]~~ 24 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 1

SR 3.3.1.7 5

This SR verifies individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall, or total, elapsed time from the point at which the parameter exceeds the analytical limit at the sensor to the point of rod insertion. Response time testing acceptance criteria for this unit are included in Reference 1.

A Note to the Surveillance indicates that neutron detectors are excluded from RPS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

Response time tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month Frequency is based on unit operating experience, which shows that random failures of

(continued)

INSERT B 3.3-29A

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION **of the resistance temperature detectors** (RTD)sensors is accomplished **by an inplace cross** calibration that compares the other sensing **elements with the recently** installed sensing element.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.7 (continued)

instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.

(5)

REFERENCES

1. UFSAR, Chapter ~~[7]~~ 7. (1)
2. UFSAR, Chapter ~~[14]~~ 15. (1)
3. 10 CFR 50.49.
4. "~~[Unit Specific Setpoint Methodology]~~." INSERT B 3.3-30A (1)
5. NUREG-0737, INSERT B 3.3-30B November 1979.
6. BAW-1893 INSERT B 3.3-30C. (2)
~~7. NRC SER for BAW-10167, Supplement 2, July 8, 1992.~~ (5)
~~8.~~ 7. BAW-10167, May 1986.
8. 10 CFR 50.36. (2)

INSERT B 3.3-30A

EDM-102, "Instrument Setpoint/Uncertainty Calculations."

INSERT B 3.3-30B

"Clarification of TMI Action Plan Requirements,"

INSERT B 3.3.30C

, "Basis for Raising Arming Threshold for Anticipating Reactor Trip on Turbine Trip," October 1985

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protect~~ion~~ive System (RPS) Manual Reactor Trip 4

BASES

BACKGROUND

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room ~~in the absence of any other trip condition~~. 2 Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switch~~es~~ contacts, one for each train. This trip is independent of the automatic trip system. 2 As shown in Figure ~~7.1~~ 7.1 UFSAR, Chapter ~~7~~ (Ref. 1), power for the ~~CONTROL ROD~~ drive (CRD) 1 breaker undervoltage coils and contactor coils comes from the reactor trip modules (RTMs). The manual trip switch~~es~~ contacts 2 are located between the RTM output and the breaker undervoltage coils. Opening of the switch~~es~~ contacts opens the lines to the breakers, tripping them. The switch~~es~~ contacts also energize the breaker shunt trip mechanisms. There is a separate switch contact 2 in series, with the output of each of the four RTMs. All switch~~es~~ contacts 2 are actuated through a mechanical linkage from a single push button.

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip Function is required as a backup to the automatic trip functions and allows operators to shut down the reactor ~~whenever any parameter is rapidly trending toward its trip setpoint~~. 2

The Manual Reactor Trip Function satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2). 2

LCO

The LCO on the RPS Manual Reactor Trip requires that the trip shall be OPERABLE whenever the reactor is critical or any time any control rod breaker is closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any ~~reactivity excursion~~ event 2 that in the operator's judgment requires protective action, even if no automatic trip condition exists.

(continued)

BASES

LCO (continued)

The Manual Reactor Trip Function is composed of four electrically independent trip switch~~es~~ contacts (2) sharing a common mechanical push button.

APPLICABILITY

The Manual Reactor Trip Function is required to be OPERABLE in MODES 1 and 2. It is also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breaker is in the closed position and if the CRD System is capable of rod withdrawal. The only safety function of the RPS is to trip the CONTROL RODS; therefore, the Manual Reactor Trip Function is not needed in MODE 3, 4, or 5 if either the reactor trip breakers are open or if the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 ~~when~~ because the CONTROL RODS are normally decoupled from the CRDs.

ACTIONS

A.1

Condition A applies when the Manual Reactor Trip Function is found inoperable. One hour is allowed to restore Function to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour Completion Time is sufficient time to correct minor problems.

B.1 and B.2

~~With the Manual Reactor Trip Function inoperable and unable to be returned to OPERABLE status within 1 hour~~ With the Required Action and associated Completion Time not met (2) in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within ~~6~~ 12 hours. The allowed Completion Time of ~~6~~ 12 (5) hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

C.1

~~With the Manual Reactor Trip Function inoperable and unable to be returned to OPERABLE status within 1 hour~~ in MODE 4

(continued)

BASES

ACTIONS

C.1 (continued)

or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that these Surveillances are only performed during a unit outage.

REFERENCES

1. UFSAR, Chapter 7. 1
2. 10 CFR 50.36. 2

B 3.3 INSTRUMENTATION

4

B 3.3.3 Reactor Protection System (RPS) – Reactor Trip Module (RTM)

XP

BASES

2

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure ~~[ ]~~ 7.1 UFSAR ~~,~~ Chapter ~~[~~7~~]~~ (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and CONTROL ROD drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS – RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip devices. 1 2

The RPS trip scheme consists of series contacts that are operated by bistables. During normal unit operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When an RTM channel trip relay de-energizes, several things occur:

a. Each of the four (4) output logic relays "informs" its associated RPS channel that a reactor trip signal has occurred in the tripped RPS channel;

b. The contacts in the trip device circuitry, powered by the tripped channel, open, but the trip device remains energized through the closed contacts from the other RTMs. (This condition exists in each RPS – RTM. Each RPS – RTM controls power to a trip device.); and

c. The contact in parallel with the channel reset switch opens and the trip is sealed in. To re-energize the channel trip relay, the channel reset switch must be depressed after the trip condition has cleared.

(continued)

BASES

BACKGROUND (continued)

When the second RPS channel senses a reactor trip condition, the output logic relays for the second channel de-energize and open contacts that supply power to the trip devices. With contacts opened by two separate RPS channels, power to the trip devices is interrupted and the CONTROL RODS fall into the core.

A minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, because the bistable relay contacts for each function are in series with the channel trip relays, two channel trips caused by different trip functions can result in a reactor trip.

4

APPLICABLE SAFETY ANALYSES

Transient and ~~A~~accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." 4

~~RTM response time is included in the overall required response time for each RPS trip and is not specified separately.~~ 6

The RTMs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2) ~~the NRC Policy Statement~~. 2

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable. ~~and reduces the reliability of the affected Functions.~~ 2

~~Four RTMs must be OPERABLE to ensure that a reactor trip would occur if needed any time the reactor is critical. OPERABILITY is defined as the RTM being~~ An OPERABLE RTM must be able to receive and interpret trip signals from its own and other RPS channels and to open its associated trip device. OPERABLE 2

The requirement of four RTMs ~~channels~~ to be OPERABLE ensures that a minimum of two RTMs ~~RPS channels~~ will remain OPERABLE if a single failure has occurred in one RTM ~~channel~~ and if a second 30

(continued)

BASES

RTM is out of service 30

LCO (continued)

2

~~channel has been bypassed for surveillance or maintenance~~. This two-out-of-four trip logic also ensures that a single ~~RPS channel~~ RTM failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2. They are also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breakers are in the closed position and the CRD System is capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur, if needed~~, any time the reactor is critical~~. This condition can exist in all of these MODES; therefore, the RTMs must be OPERABLE. 2 2

ACTIONS

A.1.1, A.1.2, and A.2

When an RTM is inoperable, the associated CRD trip breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required Action A.1.1 or Required Action A.1.2 requires this either by tripping the CRD trip breaker or by removing power to the CRD trip device. Tripping one RTM or removing power opens one set of CRD trip devices. Power to hold up CONTROL RODS is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips.

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

if two or more RTMs are inoperable or 5

Condition B applies if the ~~Required Actions~~ and associated of Condition A are not met ~~within the required~~ Completion Time in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in 2

(continued)

BASES

ACTIONS

B.1, B.2.1, and B.2.2 (continued) trip breakers

from all

which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with ~~all~~ power ~~to the~~ CRD ~~System~~ removed within ~~6~~ hours. The allowed Completion Time of ~~6~~ hours is ~~reasonable~~, based on operating experience, to reach MODE 3 from ~~full~~ power conditions in an orderly manner and without challenging unit systems. 28

12 5

5 if two or more RTMs are inoperable or

C.1 and C.2

and associated

Condition C applies if the Required Actions of Condition A are not met ~~within the required~~ Completion Time in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing ~~all~~ power ~~to the~~ CRD ~~System~~. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove ~~all~~ power ~~to the~~ CRD ~~System~~ without challenging unit systems. 2

trip breakers 28 from all

from all 28 trip breakers

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

5

~~The Note defines a channel as being OPERABLE for up to 8 hours while bypassed for Surveillance testing. The Note allows channel bypass for testing without defining it as inoperable although during this time period it cannot actuate a reactor trip. This allowance is based on the assumption of the RPS reliability analysis in BAW-10167 (Ref. 2) that 8 hours is the average time required to perform channel Surveillance. The analysis demonstrated that the 8 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified the LCO's Completion Times.~~

29

~~[Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequency is based on an approved topical report. For a licensee to use~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1 (continued)

29

[this Frequency, the licensee must justify the Frequency as required by the NRC Staff SER for the topical report.]

The SRs include performance of a CHANNEL FUNCTIONAL TEST every [45] days on a STAGGERED TEST BASIS. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals. Calculations have shown that the frequency (45 days) maintains a high level of reliability of the Reactor Trip System in BAW-10167 (Ref. 2). 1 29

29 31

INSERT B 3.3-38A

REFERENCES

1. UFSAR, Chapter [7]. 1 2

2. BAW-10167, May 1986. 29

2. 10 CFR 50.36. 2

INSERT 3.3-38A

The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event.

Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each week. Testing one RTM each week reduces the likelihood of the same systematic test errors being introduced into each redundant RTM.

4 <Except as marked>

B 3.3 INSTRUMENTATION

B 3.3.4 ~~CONTROL ROD~~ Control Rod Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protection System (RPS) contains multiple CRD trip devices: two AC trip breakers, two DC trip breaker pairs, and eight electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker and functionally in series with ~~either~~ a pair of DC breakers ~~or~~ four ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD System.

Figure ~~7.1~~ 7.1 SAR, Chapter ~~7~~ 7 (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power ~~to~~ the CRDs must be removed. Loss of power causes the ~~CRD's~~ mechanisms to release the CONTROL RODS, which then fall by gravity into the core. 1 2

Power to CRDs is supplied from two separate ~~unit~~ sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage ~~and shunt~~ trip coils are powered by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC power supplies and the regulating rod, APSR and auxiliary power supplies.

The DC power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A; the other, phase B. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls power to ~~one~~ half of the two of the four safety rod groups. The undervoltage ~~and shunt~~ trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C. The other two breakers are controlled by RPS channel D. CC

In addition to the DC power supplies, the redundant buses also supply power to the regulating rod, APSR and auxiliary power supplies. ~~These power supplies consist of ETAs that are gated on by programming lamps. Programming lamp power is controlled by contactors (E and F), which are controlled by~~

INSERT B 3.3-39A

(continued)

INSERT 3.3-39A

These power supplies contain silicon controlled rectifiers (SCRs) that are gated on and off to provide power to, and remove power from, the phases of the CRD mechanisms. The gating control signal for these SCRs is supplied through the closed contacts of the ETA relays. These contacts are referred to as E and F contactors, and are controlled by the C and D RPS channels respectively.

BASES

BACKGROUND (continued)

RPS power., One of the redundant programming lamp supplies is controlled by RPS channel C; the other, by RPS channel D. 4

The AC breaker and DC breakers are in series in one of the power supplies; whereas, the redundant AC breaker and DC breakers are in series in the other power supply to the CONTROL RODS. The logic required to cause a reactor trip is the opening of a circuit breaker in each of the redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

a. If the A AC circuit breaker opens:

   1. the input power to associated DC power supply is lost, and

   2. the SCR supply from the associated power source is lost.

b. If the D DC circuit breaker(s) and F contactors open:

   1. the output of the ~~redundant~~ DC power supply is lost ~~and the safety rods de-energize~~, and 2

   2. when the F contactor opens, ~~programming lamp~~ SCR gating power is lost ~~and the regulating rods will be de-energized~~.

c. The combination of (a) and (b) causes a reactor trip.

Any other combination of at least one circuit breaker opening in each power supply will cause a reactor trip. 4

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable lower pressure bistable relay de-energizes the channel C trip relay. When the output logic relays in channels B and C de-energize, the B and C contacts in the ~~undervoltage and E~~ trip logic of each channel's reactor trip module (RTM) open causing an undervoltage to each trip breaker. 4

(continued)

BASES

and the ETA relay contactors

4

trip

BACKGROUND (continued)

~~and E contacts de-energize~~, all ~~circuit~~ breakers open, and ~~programming lamp~~ power is removed. All rods fall into the core, resulting in a reactor trip. from all CRD mechanisms

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The control rod ~~insertion~~ position limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS ~~(except Group 8)~~ will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protect~~ion~~ re System (RPS) Instrumentation." 2 7 4

The CRD trip devices satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2). 2

LCO

required specified 2

The LCO requires all required of the specified CRD trip devices to be OPERABLE. Failure of any CRD trip device renders a portion of the RPS inoperable and reduces the reliability of the affected Functions. Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip may not ~~cannot~~ occur reliably when initiated either automatically or manually. 2 2

All required CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting ~~AC~~ power to the CRDs. Both of the ~~AC~~ breaker's diverse trip devices and the breaker itself must be functioning properly for the ~~AC~~ breaker to be OPERABLE. 2

CRD trip

Requiring all breakers and ETA relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure.

INSERT B 3.3-41A

(continued)

INSERT 3.3-41A

Both ETA relays associated with each of the three regulating rod groups and the two ETA relays associated with the auxiliary power supply must be OPERABLE to satisfy the LCO. The ETA relays associated with the APSR power supply are not required to be OPERABLE because the APSRs are not designed to fall into the core upon initiation of a reactor trip.

BASES

LCO (continued)

Requiring all devices OPERABLE also ensures that a single failure will not cause an unwanted reactor trip. 2

APPLICABILITY

The CRD trip devices shall be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

The CRD trip devices are designed to ensure that a reactor trip would occur if needed any time the reactor is critical. 2 Since this condition can exist in all of these MODES, the CRD trip devices shall be OPERABLE.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each CRD trip device.

Condition A 2

Condition A represents reduced redundancy in the CRD trip Function. Condition A applies when:

- One diverse trip Function (undervoltage or shunt trip device) is inoperable in one or more CRD trip breaker(s) for breaker pair; or 1
- One diverse trip Function is inoperable in both DC trip breakers associated with one protection channel. 4 In this case, the inoperable trip Function does not need to be the same for both breakers.

A.1 and A.2 2

If one of the diverse trip Functions on a CRD trip breaker or breaker pair becomes inoperable, actions must be taken 1 to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually tripping the inoperable CRD trip breaker or by removing power from the channel containing the inoperable CRD trip breaker. 5 Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single

(continued)

5 <Except as marked>

BASES

ACTIONS

A.1 and A.2 (continued)

failure ~~which in turn could~~ from prevent 1a ~~tripping of the reactor~~. The 48 hour Completion Time has been shown to be acceptable through operating experience. a trip 2

Condition B

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when:

One or more CRD trip breaker(s) [or breaker pair] will not function on either undervoltage or shunt trip Functions; or 2

Both diverse trip Functions are inoperable in one or ~~both DC trip breakers associated with one protection channel~~.

more trip breaker(s) or breaker pairs 2

B.1 and B.2 — 2

Required Action B.1 and Required Action B.2 are the same as Required Action A.1 and Required Action A.2, but the Completion Time is shortened. The 1 hour Completion Time allowed to trip or remove power from the CRD trip breaker allows the operator to take all the appropriate actions for the inoperable breaker and still ensures that the risk involved is acceptable.

C.1 and C.2

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of two actions to eliminate reliance on the failed ETA relay. This first option is to switch the affected control rod group to an alternate power supply. This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. The second option is to trip the corresponding AC CRD trip breaker. This results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence. 2

(continued)

(2) <Except as marked>

BASES

ACTIONS

C.1 and C.2 (continued)

The 1 hour Completion Time is sufficient to perform the Required Action.

D.1, D.2.1, and D.2.2

with ~~If the~~ Required Action~~s~~ of Condition A, B, or C ~~are~~ and associated Completion Time is not met ~~within the required Completion Time~~ in MODE 1, 2, or 3, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to ~~at least~~ MODE 3, with all CRD trip breakers open or with ~~all~~ power ~~to the~~ from all CRD ~~System~~ trip breakers removed within ~~6~~ 8 hours. The allowed Completion Time of ~~6~~ 8 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. 28 12 5

E.1 and E.2

with ~~If~~ the Required Action~~s~~ of Condition A, B, or C ~~are~~ and associated Completion Time is not met ~~within the required Completion Time~~ in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or ~~all~~ power ~~to the~~ from all CRD ~~System~~ trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove ~~all~~ power ~~to the~~ from all CRD ~~System~~ trip breakers without challenging unit systems. 28

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 31 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the ~~AC~~ trip breakers. The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event.

REFERENCES

1. ~~U~~FSAR, Chapter ~~[7]~~ 7. (2) (1)
2. 10 CFR 50.36. (2)

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safeguards Protective ~~Safety Feature Actuation~~ System (ESPS ~~ESFAS~~) Analog Instrumentation

BASES

BACKGROUND

The ESPS ~~ESFAS~~ initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and reactor coolant pressure boundary and to mitigate accidents.

ESPS ~~ESFAS~~ actuates the following systems:

- High pressure injection (HPI) Actuation;
- Low pressure injection (LPI) Actuation;
- Reactor building (RB) Cooling;
- Penetration room ventilation;
- ~~RB Spray;~~
- RB Isolation; and
- Keowee Hydro Unit Emergency Emergency diesel generator (EDG) Start.

ESFAS also provides a signal to the Emergency Feedwater Isolation and Control (EFIC) System. This signal initiates emergency feedwater (EFW) when HPI is initiated. 8

The ESPS ~~ESFAS~~ operates in a distributed manner to initiate the appropriate systems. The analog ESPS ~~ESFAS~~ does ~~this~~ by determining the need for actuation in each of three channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to digital automatic actuation logics, which perform the two-out-of-three logic to determine the actuation of each end device. Each end device has its own automatic actuation logic, although all automatic actuation logic channels take their signals from the same bistable ~~point~~ in each channel for each Parameter.

Four Parameters are used for actuation:

- Low Reactor Coolant System (RCS) Pressure;
- Low Low RCS Pressure;
- High RB Pressure; and

(continued)

BASES

BACKGROUND (continued)

- High High RB Pressure.

LCO 3.3.5 covers only the analog instrumentation channels that measure these Parameters. These channels include all intervening equipment necessary to produce actuation before the measured process Parameter exceeds the limits assumed by the accident analysis. This includes sensors, bistable devices, operational bypass circuitry, block timers, and output relays. LCO 3.3.6, "Engineered Safeguards Protective ~~Safety Feature Actuation~~ System (~~ESFAS~~) Manual Initiation," and LCO 3.3.7, "Engineered Safeguards ~~Safety Feature Actuation~~ System (~~ESFAS~~) ESPS Digital Automatic Actuation Logic Channels," provide requirements on the manual initiation and digital automatic actuation logic Functions.

The ESPS ~~ESFAS~~ contains ~~consists of~~ three analog ~~protection~~ channels. Each analog channel provides input to digital logic channels ~~logics~~ that initiate equipment with a two-out-of-three logic on each ~~component~~. Each analog ~~protection~~ channel includes digital ~~bistable~~ inputs from one instrumentation channel of Low RCS ~~RB~~ Pressure, Low Low RCS Pressure, High RB Pressure, and High High RB Pressure. 2 Digital automatic actuation logic channels combine the three analog ~~protection~~ channel trips in each train to actuate the individual Engineered Safeguards ~~Safety Feature~~ (ESP) components needed to initiate each ESP System. 10 Figure ~~[ ]~~ 7.5, FSAR, Chapter ~~[7]~~ 7 1 (Ref. 1), illustrates how analog instrumentation channel trips combine to cause digital logic ~~protection~~ channel trips. 2

The RCS pressure sensors are common to both trains. Isolation is provided via separate bistables for each train. Separate RB pressure sensors are used for the high and high high pressure Functions in each train, and separate sensors are used for each train.

The following matrix identifies the analog instrumentation ~~(measurement)~~ channels and the Digital Automatic Actuation Logic Channels ~~Function~~ actuated by each. 2

(continued)

BASES

BACKGROUND (continued)

INSERT B 3.3-47A

9

PARAMETER	LOW RCS PRESSURE	LOW LOW RCS PRESSURE	HIGH RB PRESSURE	HIGH HIGH RB PRESSURE
HPI	X	X	X	
LPI		X		X
RB Cooling	X	X	X	(b)
RB Spray	(b)			
RB Isolation(a)	X	X	X	
EDG Start	X	X	X	
Control Room Isolation			X	

(a) Only isolates systems not required for RB or RCS heat removal.

(b) Actuates on High High RB Pressure coincident with HPI actuation.

8

~~Engineered safeguards bus undervoltage will also sequence on the HPI loads started by the HPI block timers. However, HPI will not occur unless the ESFAS HPI signal is also present. LCO 3.3.8, "Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)," contains the requirements for the undervoltage channels.~~

The ESP equipment is 4 generally 2 divided between the two redundant digital actuation ~~trains A and B~~. ~~The division of the equipment between the two actuation trains~~ logic channels is based on the equipment redundancy and function and is accomplished in such a manner that the failure of one of the logic actuation channels and the related safeguards equipment will not inhibit the overall ESP Functions. ~~Where a motor operated or a solenoid operated valve is driven by either of two matrices, one is from actuation channel A and one from actuation channel B.~~ Redundant ESP pumps are controlled from separate and independent digital actuation logic channels, with the exception of HPI B pump which is actuated by both.

The actuation of ESP equipment is also available by manual actuation switches located on the control room console or ES panel.

(continued)

INSERT B3.3-47A

Digital Logic Channels	Actuated Systems/ Functions	RCS PRESS LOW	RCS PRESS LOW LOW	RB PRESS HIGH	RB PRESS HIGH HIGH
1 and 2	HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	X		X	
3 and 4	LPI and RB Essential isolation		X	X	
5 and 6	RB Cooling, RB Essential isolation, and Penetration Room Vent.			X	
7 and 8	RB Spray				X

BASES

BACKGROUND (continued)

The ESPS ~~ESFAS,~~ in conjunction with the actuated equipment, provides protective functions necessary to mitigate ~~Design Basis~~ Accidents ~~(DBAs)~~, specifically the loss of coolant accident (LOCA) and main steam line break (SLB) events. The ESPS ~~ESFAS~~ relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems of LCO 3.3.7. (M)

Engineered ~~Safety Feature Actuation~~ Safeguards Protective System Bypasses

No provisions are made for maintenance bypass of ESPS ~~ESFAS~~ instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow ~~reactor~~ unit shutdown without spurious ESFAS actuation. (2)

The ESPS ~~ESFAS~~ RCS pressure instrumentation channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low and low low pressure trips are required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded.

(2) INSERT B 3.3-48A

(8) Each High RB Pressure channel may be manually bypassed after the other two channels in the Parameter have tripped. The manual bypass allows operators to take manual control of ESF Functions after initiation to allow recovery actions. The bypass may be manually removed and is automatically removed when RB pressure returns to below the trip setpoint.

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protect~~ion~~ive System (RPS). Each of the pressure signals generated by these transmitters is monitored by four bistables to provide two trip signals, at ≥ 1500 psig and ≥ 500 psig, (2) and two bypass permissive signals, at ≤ 17~~00~~50 psig and ≤ 900 psig. (2) (5)

The outputs of the three bistables, associated with the low RCS pressure, ≥ 1500 psig, trip drive relays in two sets

(continued)

INSERT B 3.3-48A

This bypass provides an operational provision only outside the Applicability for this parameter, and provides no safety function.

BASES

BACKGROUND

Reactor Coolant System Pressure (continued)

(actuation channels ~~A and B~~) of identical and independent channels. These two sets of HPI channels each use ~~three logic channels used in~~ two-out-of-three coincidence networks for HPI Actuation. The outputs of the three bistables associated with the Low Low RCS Pressure~~, 500 psig,~~ trip drive relays in two sets ~~(actuation channels A and B)~~ of identical and independent channels. These two sets of LPI channels each ~~use three logic channels used in~~ two-out-of-three coincidence networks for LPI Actuation. The outputs of the three Low Low RCS Pressure bistables also trip the drive relays in the corresponding HPI Actuation channel as previously described. (2)

(a) use a (2)

Reactor Building Pressure (8)

INSERT B3.3-49A

RB pressure inputs to the ESFAS are provided by 12 pressure switches. Six pressure switches are used for the High RB Pressure Parameter, and six pressure switches are used for the High High Pressure Parameter.

The output contacts of six High RB Pressure switches are used in two sets of identical and independent actuation trains. These two trains each use three logic channels. The outputs of these channels are used in two-out-of-three coincidence networks. The output contacts of the six RB pressure switches also trip the drive relays in the corresponding HPI and LPI Actuation channels as previously described.

The output contacts of six High High RB Pressure switches are used in two sets of identical and independent actuation trains. These two trains each use three logic channels (RB4, RB5, and RB6). The outputs of these channels are used in two-out-of-three coincident networks for RB Spray Actuation. Each high high pressure train actuates one RB Spray train when the High High RB signal and the HPI signal are coincident in that train.

Trip Setpoints and Allowable Values

Trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted

(continued)

INSERT B3.3-49A

There are three Reactor Building pressure sensors. The output of each sensor terminates in an input isolation amplifier, which provides individually isolated outputs. One isolated output of each pressure measurement goes to the unit computer for monitoring. One output of each pressure measurement goes to a bistable which initiates action when its high building pressure trip point is exceeded. Each input isolation amplifier module contains an analog meter for indicating the measured pressure. Each of the three bistables has contact outputs that are combined in series with the output of the High and Low Pressure Injection System bistables as previously described.

The outputs of the three bistables are brought together in two identical two-out-of-three coincidence logics which provide two ESPS channels. Either of the two channels is independently capable of initiating the required protective action.

The ESPS channels of the Reactor Building Spray System are formed by two identical two-out-of-three logic networks with the active elements originating in six Reactor Building pressure sensing pressure switches.

Three independent pressure switches containing normally open contacts from one protective channel's two-out-of-three logic inputs. Three other identical pressure switches from the two-out-of-three logic inputs of the second protective channel. Either of the two protective channels is capable of initiating the required protective action.

BACKGROUND

Trip Setpoints and Allowable Values (continued)

when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy]). (2)

The trip setpoints used in the bistables are based on the analytical limits stated in Figure [ ], FSAR, Chapter [7] (Ref. 1). The selection of these trip setpoints is such Selected that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment induced errors for those ESPS ~~ESFAS~~ channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2), the Allowable Values specified in Table 3.3.5-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the (2) methodology used to calculate the trip setpoints, including their explicit (2) uncertainties, is provided in the Reference 3 ~~"Unit Specific Setpoint Methodology" (Ref. 3).~~ The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. (8) A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Setpoints, in accordance with the Allowable Values, ensure that the consequences of accidents ~~DBAs~~ will be acceptable, providing the unit is operated from within the LCOs at the onset of the accident ~~DBA~~ and the equipment functions as designed.

Each channel can be tested online to verify that the setpoint accuracy is within the specified allowance requirements of Reference 3. (2) Once a designated channel is taken out of service for testing, a simulated signal ~~is~~ may be injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. (2)

The Allowable Values listed in Table 3.3.5-1 are based on the methodology described in FSAR, Chapter [14] (Ref. 4), which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are (8)

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued) 8

factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

[Reviewer's Note: The ESFAS LCOs in the BWOG Standard Technical Specifications are based on a system representative of the Crystal River Unit 3 design.] As discussed earlier, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple components are affected by each instrumentation channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflects the described logic arrangement by identifying instrumentation requirements on an instrumentation channel rather than on a protective function basis. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the different impact each has on ESFAS OPERABILITY. 2

APPLICABLE SAFETY ANALYSES

The following ESPS 4 ~~ESFAS~~ Functions have been assumed within the accident analyses.

High Pressure Injection

The ~~ESFAS~~ ESPS actuation of HPI has been assumed for core cooling in the LOCA analysis and is credited with boron addition in the M 4 SLB analysis.

4 ESPS

Low Pressure Injection

The ~~ESFAS~~ ESPS actuation of LPI has been assumed for large break LOCAs.

(continued)

4 (Except as marked)

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ~~ESFAS~~ ESPS actuation of the RB coolers and RB Spray have been credited in RB analysis for LOCAs, both for RB performance and equipment environmental qualification pressure and temperature envelope definition. Accident dose calculations have credited RB Isolation and RB Spray.

INSERT B 3.3-52A — 2

~~Emergency Diesel Generator~~ Keowee Hydro Unit Emergency Start

The ~~ESFAS~~ ESPS initiated ~~EDG~~ Start has been ~~assumed in the LOCA analysis~~ included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small and large break LOCA analyses assume a conservative ~~35~~ 48 second delay time for the actuation of HPI and LPI in ~~F~~SAR, Chapter ~~14~~ 15 (Ref. 4). This delay time includes allowances for ~~EDG~~ Keowee Hydro Unit starting, ~~EDG loading~~, Emergency Core Cooling Systems (ECCS) pump starts, and valve openings. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed. ~~Typical values used in the analysis are 35 seconds for RB Cooling, 60 seconds for RB Isolation, and 56 seconds for RB Spray.~~ 2

Accident analyses rely on automatic ~~ESFAS~~ ESPS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, ~~SLB, and Feedwater line break~~ events that result in RCS inventory reduction ~~or severe loss of RCS cooling~~. M

The ~~ESFAS~~ ESPS channels satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 5). 2

LCO

The LCO requires three analog channels of ~~ESFAS~~ ESPS instrumentation for each Parameter in Table 3.3.5-1 to be OPERABLE in each digital automatic actuation logic channel ~~ESFAS train~~. Failure of any instrument renders the affected analog channel(s) inoperable and reduces the reliability of the affected Functions.

(continued)

INSERT B 3.3-52A

Penetration Room Ventilation Actuation

The ESPS actuation of the penetration room ventilation system has been assumed for LOCAs. Accident dose calculations have credited penetration room ventilation.

BASES

LCO (continued)

Only the Allowable Value is specified for each ~~ESFAS~~ ESPS 4 Function in the LCO. Nominal trip setpoints are specified in the ~~unit specific~~ setpoint calculations. The nominal trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the ~~unit specific~~ setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip Parameter. These uncertainties are defined in ~~the "Unit Specific Setpoint Methodology" (Ref. 3)~~ Reference 3. 2 2 2

The Allowable Values for bypass removal functions are stated in the Applicable MODES or Other Specified Condition column of Table 3.3.5-1.

Three ~~ESFAS~~ ESPS 4 analog 4 instrumentation channels shall be OPERABLE ~~in each ESFAS train~~ 10 to ensure that a single failure in one analog 4 channel will not result in loss of the ability to automatically actuate the required safety systems.

The bases for the LCO on ~~ESFAS~~ Parameters include the following.

~~Reactor Coolant System Pressure~~ 2

Three analog 4 channels ~~each~~ of RCS Pressure - Low ~~and RCS Pressure - Low Low~~ and RB Pressure-High and RB Pressure-High High 8 are required OPERABLE ~~in each train~~ 10. Each analog 4 channel includes a sensor, trip bistable, bypass bistable, ~~bypass relays~~, output relays, ~~and block timers. The analog portion of each pressure channel is common to both trains of both RCS Pressure Parameters. Therefore, failure of one analog channel renders one channel of the low pressure and low low pressure Functions in each train inoperable. The bistable portions of the channels are Function and train specific. Therefore, a bistable failure renders only one Function in one train inoperable.~~ 8 Failure of a bypass analog 4 bistable or bypass circuitry, such that ~~a trip~~ 2 channel cannot be bypassed, does not render the analog channel inoperable. Since the analog channel is still capable of performing its safety function, i.e., this is not a safety related bypass function. ~~Output relays and block timer relays are train specific but may be shared among Parameters. Therefore, output or block~~ 10

(continued)

BASES

LCO

~~Reactor Coolant System Pressure (continued)~~

~~timer relay failure renders all affected Functions in one train inoperable.~~ 10

1. Reactor Coolant System Pressure - Low Setpoint

~~The RCS Pressure - Low Setpoint is based on HPI actuation for small break LOCAs. The setpoint ensures that the HPI will be actuated at a pressure greater than or equal to the value assumed in accident analyses plus the instrument uncertainties. The maximum value assumed for the setpoint of the RCS Pressure - Low trip of HPI in safety analyses is 1480 psig. The setpoint for the low RCS and Allowable Value of ≥ [1600] psig for the low pressure Parameter is selected to ensure actuation occurs when actual RCS pressure is above 1480 psig. The RCS Pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint Allowable Value accounts for severe environment induced errors.~~

~~To ensure the RCS Pressure - Low trip is not bypassed when required to be OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an Allowable Value of ≤ [1800] psig. The bypass removal does not need to function for accidents initiated from RCS Pressures below the bypass removal setpoint. Therefore, the bypass removal setpoint Allowable Value need not account for severe environment induced errors.~~

2. Reactor Coolant System Pressure - Low Low Setpoint

~~The RCS Pressure - Low Low Setpoint LPI actuation occurs in sufficient time to ensure LPI flow prior to the emptying of the core flood tanks during a large break LOCA. The Allowable Value of ≥ [400] psig ensures sufficient overlap of the core flood tank flow and the LPI flow to keep the reactor vessel downcomer full during a large break LOCA. The RCS Pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint Allowable Value accounts for severe environment induced errors.~~

8

(continued)

BASES

LCO 8

2. Reactor Coolant System Pressure - Low Low Setpoint (continued)

To ensure the RCS Pressure - Low Low trip is not bypassed when assumed OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an Allowable Value of ≤ [900] psig. The bypass removal does not need to function for accidents initiated by RCS Pressure below the bypass removal setpoint. Therefore, the bypass removal setpoint Allowable Value need not account for severe environment induced errors.

Reactor Building Pressure

Three channels each of RCS Pressure - Low and RB Pressure - High are required to be OPERABLE in each train. Each channel includes a pressure switch, bypass relays, and output relays. The high pressure channels also include block timers. Each pressure switch is Function and train specific, so there are 12 pressure switches total. Therefore, a pressure switch renders only one Function in one train inoperable. Output relays and block timer relays are train specific but may be shared among Parameters. Therefore, output or block timer relay failure renders all affected Functions in one train inoperable.

The RB Pressure switches may be subjected to high radiation conditions during the accidents that they are intended to mitigate. The sensor portion of the switches is also exposed to the steam environment present in the RB following a LOCA or high energy line break. Therefore, the trip setpoint Allowable Value accounts for measurement errors induced by these environments.

1. Reactor Building Pressure - High Setpoint

The RB Pressure - High Setpoint Allowable Value ≤ [5] psig was selected to be low enough to detect a rise in RB Pressure that would occur due to a small break LOCA, thus ensuring that the RB high pressure actuation of the safety systems will occur for a wide spectrum of break sizes. The trip setpoint also causes the RB coolers to shift to emergency mode to prevent damage to the cooler fans due to the increase

(continued)

BASES

LCO

1. Reactor Building Pressure - High Setpoint (continued) 8

in the density of the air steam mixture present in the containment following a LOCA.

2. Reactor Building Pressure - High High Setpoint

The RB Pressure - High High Setpoint Allowable Value ≤ [30] psig was chosen to be high enough to avoid actuation during an SLB, but also low enough to ensure a timely actuation during a large break LOCA.

APPLICABILITY

Three analog (4) channels of ~~ESFAS~~ ESPS instrumentation for each of the following Parameters (2) ~~listed next~~ (2) shall be OPERABLE ~~in each ESFAS train~~ (5).

1. Reactor Coolant System Pressure - Low ~~Setpoint~~ (2)

The RCS Pressure - Low ~~Setpoint~~ actuation Parameter shall be OPERABLE during operation at or above ~~1800~~ 1750 (4) psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below ~~1800~~ psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. ~~Plant~~ RCS (2) pressure and temperature

(continued)

BASES

APPLICABILITY

1. Reactor Coolant System Pressure - Low ~~Setpoint~~ 2
(continued)

are very low, and many ~~ESF~~ ES 4 components are administratively ~~locked out~~ controlled 2 or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

2. Reactor Coolant System Pressure - Low Low ~~Setpoint~~ 2

The RCS Pressure - Low Low ~~Setpoint~~ actuation Parameter shall be OPERABLE during operation above ~~[900]~~ psig. 1 This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below ~~[900]~~ psig, the low low RCS 1 Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety system actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other 2 equipment to mitigate the consequences of an abnormal condition or accident. ~~Plant~~ RCS pressure and temperature are very low, and many ES~~F~~ 4 components are administratively ~~locked out~~ controlled 2 or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

(continued)

BASES

APPLICABILITY (continued)

3, 4. Reactor Building Pressure - High and Reactor Building Pressure - High High ~~Setpoints~~ 2

The RB Pressure - High and RB Pressure - High High actuation Functions of ESPS 4 ~~ESFAS~~ shall be OPERABLE in MODES 1, 2, 3, and 4 when the potential for a HELB exists. In MODES 5 and 6, the unit conditions are such that there is insufficient energy in the primary and secondary systems to raise the containment pressure to either the RB Pressure - High or RB Pressure - High High actuation 2 ~~Setpoints~~. Furthermore, in MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. ~~Plant~~ RCS 2 pressure and temperature are very low and many ESP components are administratively controlled ~~locked out~~ or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

ACTIONS

Required Actions A and B apply to all ESPS 4 ~~ESFAS~~ analog 4 instrumentation Parameters listed in Table 3.3.5-1.

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Parameter.

If ~~a~~ an analog 4 channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or ~~ESFAS~~ ESPS 4 bistable is found inoperable, then all affected functions provided by that analog 4 channel should be declared inoperable and the unit must enter the Conditions for the particular protective 4 ~~protection~~ Parameter affected.

~~When the number of inoperable channels in a trip Parameter exceeds those specified, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 shall be immediately entered if applicable in the current MODE of operation.~~ 5

(continued)

BASES

ACTIONS (continued)

A.1

Condition A applies when one analog channel becomes inoperable in one or more Parameters. If one ~~ESFAS~~ ESPS analog instrument channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another analog channel were to fail, the ~~ESFAS~~ ESPS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays ~~and block timers~~ are tripped. This can normally be accomplished by tripping the affected bistables ~~or tripping the individual output relays and block timers. [At this unit, the specific output relays associated with each ESFAS instrumentation channel are listed in the following document:]~~ 8

The 1 hour Completion Time is sufficient time to perform the Required Action.

B.1, B.2.1, B.2.2, and B.2.3

Condition B applies when the Required Action ~~A.1 is~~ and associated Completion Time of Condition A are not met ~~within the required Completion Time~~ or when one or more Parameters have two or more inoperable analog channels. If Condition B applies, ~~Required Action A.1 cannot be met within the required Completion Time,~~ the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within ~~6~~ 12 hours and, for the RCS Pressure – Low ~~Setpoint~~ Parameter, to < ~~[1800]~~ 1750 psig, for the RCS Pressure – Low Low ~~Setpoint~~ Parameter, to < ~~[900]~~ psig, and for the RB Pressure – High ~~Setpoint~~ Parameter and RB Pressure – High High ~~Setpoint~~ Parameter, to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

~~All~~ The ~~ESFAS~~ Parameters listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION ~~, and response time testing~~. The operational bypasses associated with each ~~ESFAS~~ RCS Pressure instrumentation channel are also subject to these SRs ~~to ensure~~ OPERABILITY of the ~~ESFAS~~ ESPS instrumentation channel.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1

Performance of the CHANNEL CHECK every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that analog ~~instrument~~ channels monitoring the same parameter should read approximately the same value. Significant deviations between the two analog ~~instrument~~ channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined ~~by the unit staff~~, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. ~~If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.~~

The Frequency, equivalent to ~~about once~~ every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel operability during normal operational use of the displays associated with the LCO's required channels.

SR 3.3.5.2

A Note permits delaying entry into applicable Conditions and Required Actions ~~defines a channel as being OPERABLE~~ for up to 8 hours while bypassed for Surveillance testing provided the remaining two ESPS analog instrument ~~ESFAS~~ channels are OPERABLE or tripped. The Note allows channel bypass for testing without entering the Conditions and Required Actions ~~defining it~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2 (continued)

12 as ~~inoperable~~, although during this time period it cannot initiate ~~ESFAS~~. This allowance is based on the inability to perform the Surveillance in the time permitted by the Required Actions. Eight hours is ~~the average~~ sufficient time ~~required~~ to perform the Surveillance. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing. 2

A CHANNEL FUNCTIONAL TEST is performed on each required ~~ESFAS~~ ESPS analog channel to ensure the entire channel, including the bypass function, will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis. 8 4 4

The Frequency of 31 days is based on ~~unit~~ operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the analog instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION ~~shall find~~ assures that measurement errors and bistable setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the ~~unit specific~~ setpoint analysis. 4 2

~~This~~ Frequency is justified by the assumption of an ~~[18]~~ month calibration interval to determine the magnitude of equipment drift in the setpoint analysis. 1

5

~~SR 3.3.5.4~~

~~SR 3.3.5.4 ensures that the ESFAS actuation channel response times are less than or equal to the maximum times assumed in the accident analysis. The response time values are the~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.4 (continued)

maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. Response time testing acceptance criteria for this unit are included in Reference 1. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the actuation setpoint value at the sensor to the point at which the end device is actuated. Thus, this SR encompasses the automatic actuation logic components covered by LCO 3.3.7 and the operation of the mechanical ESF components.

Response time tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The 18 month test Frequency is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation but not channel failure are infrequent occurrences.

REFERENCES

1. UFSAR, Chapter ~~[7]~~ 9.
2. 10 CFR 50.49.
3. ~~"Unit Specific Setpoint Methodology."~~ INSERT B 3.3-62A
4. UFSAR, Chapter ~~[14]~~ 15.
5. 10 CFR 50.36.

INSERT B 3.3-62A

EDM-102, "Instrument Setpoint/Uncertainty Calculations."

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered Safeguards Protective ~~Safety Feature Actuation~~ System (ESPS ~~ESFAS~~) Manual Initiation

BASES

BACKGROUND

The ESPS ~~ESFAS~~ manual initiation capability allows the operator to actuate ~~ESFAS~~ Functions from the main control room in the absence of any other initiation condition. Manually actuated Functions include High Pressure Injection, Low Pressure Injection, Reactor Building (RB) Cooling, RB Spray, RB Isolation, and Control Room Isolation. This ~~ESFAS~~ ESPS manual initiation capability is provided in the event the operator determines that an ~~ESFAS~~ ESPS Function is needed and has not been automatically actuated. Furthermore, the ~~ESFAS~~ ESPS manual initiation capability allows operators to rapidly initiate Engineered Safeguards ~~Safety Feature~~ (ES ~~F~~) Functions. ~~if the trend of unit parameters indicates that ESF actuation will be needed.~~ ⑧ ②

LCO 3.3.6 covers only the ESPS system level manual initiation of these Functions. LCO 3.3.5, "Engineered Safeguards Protective ~~Safety Feature Actuation~~ System (~~ESFAS~~) Analog Instrumentation," and LCO 3.3.7, "Engineered Safeguards Protective ~~Safety Feature Actuation~~ System (~~ESFAS~~) ESPS Digital Automatic Actuation Logic Channels," provide requirements on the portions of the ~~ESFAS~~ ESPS that automatically initiate the Functions described earlier.

The ~~ESFAS~~ ESPS manual initiation Function relies on the OPERABILITY of the digital ~~automatic~~ actuation logic channels (LCO 3.3.7) ~~for each component~~ to perform the actuation of the systems. A manual trip push button is provided on the ~~ESF panel of the~~ control room console for each of the ~~levels of protection for each actuation~~ digital automatic actuation logic channels. Operation of the push button energizes relays whose contacts perform a logical "OR" function with the ~~matrices of the~~ automatic actuation. ~~except for the matrices which are part of the ESF buses loading sequence. Manual actuation of the ESF buses loading sequence is made by de-energizing the timed output relays. The power supply for the manual trip relays is taken from the station batteries. Different batteries are used for the two actuations.~~ ⑧

The ESPS ~~ESFAS~~ manual initiation channel is defined as the instrumentation between the console switch and the digital automatic actuation logic channel, which actuates the end devices. Other means of manual initiation, such as controls for individual ES ~~F~~ devices, may be available in the control room and other

(continued)

4 <Except as marked>

BASES

BACKGROUND (continued)

unit locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.

APPLICABLE SAFETY ANALYSES

The ~~ESFAS~~ ESPS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate ~~Design Basis~~ Accidents, specifically, the loss of coolant accident and steam line break events.

The ~~ESFAS~~ ESPS manual initiation ensures that the control room operator can rapidly initiate ESP Functions ~~at any time~~. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ~~ESFAS~~ ESPS whenever any parameter is rapidly trending toward its trip setpoint. ~~Furthermore, the ESFAS manual initiation may be specified in operating procedures for verification that ESF systems are running.~~ 18

The ~~ESFAS~~ ESPS manual initiation functions satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 1). 2

LCO

Two ~~ESFAS~~ ESPS manual initiation channels of each ~~ESFAS~~ ESPS Function shall be OPERABLE whenever conditions exist that could require ESP protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ~~ESFAS~~ ESPS Function. The ~~ESFAS~~ ESPS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

8 INSERT B 3.3-64A

APPLICABILITY

The ~~ESFAS~~ ESPS manual initiation ~~Functions~~ MODES 3 and Functions shall be OPERABLE in MODES ~~1, 2, and 3,~~ and in ~~MODE~~ 4 when the associated engineered safeguard equipment is required to be OPERABLE. The manual initiation channels are required because ESP Functions are designed to provide protection in these MODES. ~~In MODES 5 and 6, ESFAS~~ ESPS initiates systems that are either reconfigured ~~or disabled~~ for ~~shutdown cooling~~ decay heat removal operation or disabled while in MODES 5 and 6. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and 2

(continued)

INSERT B3.3-64A

The required Function is provided by two associated channels as indicated in the following table:

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling, RB Essential isolation, and Penetration Room Vent.	5 & 6
RB Spray	7 & 8

4 <Except as marked>

BASES

APPLICABILITY (continued)

to respond by manually operating the ESP components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ~~ESFAS~~ ESPS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more ~~ESFAS~~ ESPS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. The Completion Time of 72 hours is based on ~~unit~~ operating experience and administrative controls, which provide alternative means of ~~ESFAS~~ ESPS Function initiation via individual component controls. The 72 hour Completion Time is generally consistent with the allowed outage time for the safety systems actuated by ~~ESFAS~~ ESPS. 2

2

B.1 and B.2

Required Action B.1 and Required Action B.2 apply if Required Action A.1 cannot be met within the required Completion Time. ~~If~~ with the Required Action ~~A.1 cannot be met within the required~~ and associated Completion Time ~~, the~~ not met unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within ~~6~~ 12 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems. 2 2 5

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires ESPS the performance of a CHANNEL FUNCTIONAL TEST of the ~~ESFAS~~ manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the ~~end device (i.e., pump, valves, etc.)~~ automatic actuation logic channels. The ~~[~~18~~]~~ month Frequency is based on the need to perform this Surveillance 8 1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the ~~[~~18~~]~~ month Frequency. (1)

REFERENCES

~~None~~ 1. 10 CFR 50.36. (2)

4 <Except as marked>

B 3.3 INSTRUMENTATION

B 3.3.7 Engineered Safeguards Protective ~~Safety Feature Actuation~~ System (ESPS ~~ESFAS~~) Digital Automatic Actuation Logic Channels

BASES

BACKGROUND

The digital automatic actuation logic channels of ~~ESFAS~~ ESPS are defined as the instrumentation ~~logic~~ ~~between~~ through the buffers of the ESPS analog instrument ~~sensing~~ channels ~~and the~~ from the controllers that actuate ~~ESFAS~~ ESPS equipment. Each unit of the components actuated by the ~~ESFAS~~ functions ~~has an~~ associated ~~automatic actuation logic matrix~~ digital automatic actuation logic channels. If two-out-of-three ~~ESFAS~~ ESPS analog instrumentation channels indicate a trip, or ~~system~~ channel level manual initiation occurs, the digital automatic actuation logic channels is activated and the associated ~~component~~ equipment is actuated. The purpose of requiring OPERABILITY of the ~~ESFAS~~ ESPS digital automatic actuation logic channels is to ensure that the Functions of the ~~ESFAS~~ ESPS with one or more digital automatic actuation logic channels can be automatically initiated in the event of an accident. Automatic actuation of some Functions is necessary to prevent the unit from exceeding the Emergency Core Cooling Systems (ECCS) limits in 10 CFR 50.46 (Ref. 1). It should be noted that OPERABLE digital automatic actuation logic channels alone will not ensure that each Function can be activated; the analog instrumentation channels and actuated equipment associated with each Function must also be OPERABLE to ensure that the Functions can be automatically initiated during an accident.

LCO 3.3.7 covers only the digital ESPS automatic actuation logic channels that initiates these Functions. LCO 3.3.5, "Engineered Safeguards Protective ~~Safety Feature Actuation~~ System (~~ESFAS~~) Analog ~~Instrumentation,~~ and LCO 3.3.6, "Engineered ~~Safety Feature Actuation~~ System (~~ESFAS~~) Manual Initiation," provide requirements on the analog instrumentation and manual initiation channels that input to the digital automatic actuation logic channels.

The ESPS ~~ESFAS~~, in conjunction with the actuated equipment, provides protective functions necessary to mitigate ~~Design Basis~~ Accidents ~~(DBAs)~~, specifically, the loss of coolant accident (LOCA) and main steam line break (MSLB) events. The ESPS ~~ESFAS~~ relies on the OPERABILITY of the automatic actuation logic for each component to perform the actuation of the selected systems.

The small and large break LOCA analyses assume a conservative 48 ~~35~~ second delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in

(continued)

UFSAR, Chapter 15

Keowee Hydro Unit startup and

BASES

BACKGROUND (continued)

~~BAW-10103A, Rev. 3~~ (Ref. 2). This delay time includes allowances for ~~emergency diesel generator (EDG) starts, EDG~~ loading, ECCS pump starts, and valve openings. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

~~Typical values used in the analyses are 35 seconds for RB Cooling, 60 seconds for RB Isolation, and 58 seconds for RB Spray.~~ 4

4

The ~~ESFAS~~ ESPS automatic initiation of Engineered ~~Safety Feature~~ Safeguards (ES~~F~~) Functions to mitigate accident conditions is assumed in the DBA accident analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, and RB Isolation.

~~The ESFAS LCOs in the BWOG Standard Technical Specifications (STS) are based on a system representative of the Crystal River Unit 3 design. As discussed earlier, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple ESF components are affected by a measurement channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflect the described logic arrangement by linking actions for automatic actuation logic failures directly to the actions for the affected ESF component. The overall philosophy is that if an automatic actuation logic fails, the affected component is put into its engineered safeguard configuration. This action eliminates the need for the automatic actuation logic. If the affected component cannot be placed in its engineered safeguard configuration, actions are taken to address the inoperability of the supported system component. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the different impact each has on ESFAS OPERABILITY.~~ 4

(continued)

4 <Except as marked>

BASES (continued)

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ~~ESFAS~~ ESPS actuation for protection of the core and RB and for limiting off site dose levels following an accident. ~~These include LOCA, SLB, and feedwater line break events that result in Reactor Coolant System (RCS) inventory reduction or severe loss of RCS cooling.~~ 4 The digital automatic actuation logic is an integral part of the ~~ESFAS.~~ ESPS.

The ~~ESFAS~~ ESPS digital automatic actuation logic channels satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 2

LCO

The digital automatic actuation logic ~~matrix for each component actuated by the ESFAS is~~ channels are required to be OPERABLE whenever conditions exist that could require ES~~F~~P protection of the reactor or the RB. This ensures automatic initiation of the ES~~F~~P required to mitigate the consequences of accidents. 5

8 INSERT B 3.3-69A

APPLICABILITY

The digital automatic actuation logic ~~Function~~ channels shall be OPERABLE in MODES ~~1,~~ ~~2, and~~ ~~3,~~ MODES 3 and ~~MODE~~ 4 when the associated engineered safeguard equipment is required to be OPERABLE because ES~~F~~P Functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by the ~~ESFAS~~ ESPS are either reconfigured ~~or disabled~~ for ~~shutdown cooling~~ decay heat removal operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and respond by manually operating the ES~~F~~P components, if required. 5 2

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ~~ESFAS~~ ESPS digital automatic actuation logic ~~matrix~~ channel. 5

A.1 and A.2

When one or more digital automatic actuation logic ~~matrices~~ channels are inoperable, the associated component(s) can be placed in ~~its~~ their engineered safeguard configuration. Required Action A.1 is 5 2

(continued)

INSERT B3.3-69A

The required Function is provided by two associated digital channels as indicated in the following table:

Function	Associated Channels
HPI and RB Non-Essential Isolation, Keowee Emergency Start, Load Shed and Standby Breaker Input, and Keowee Standby Bus Feeder Breaker Input	1 & 2
LPI and RB Essential isolation	3 & 4
RB Cooling, RB Essential isolation, and Penetration Room Vent.	5 & 6
RB Spray	7 & 8

BASES

ACTIONS

A.1 and A.2 (continued)

equivalent to the digital automatic actuation logic channel performing its safety function ahead of time. In some cases, placing the component in its engineered safeguard configuration would violate unit safety or operational considerations. In these cases, the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its engineered safeguard configuration include, but are not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, or isolation of fluid systems that are normally functioning. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component. declaring associated components

Required Action A.2 requires ~~entry into the Required Actions of the affected supported systems~~ declaring associated components inoperable, since the true effect of digital automatic actuation logic channel failure is inoperability of the supported system. ~~The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.~~

INSERT B 3.3-70A

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 ~~is the performance of~~ a CHANNEL FUNCTIONAL TEST on a 31 day ~~STAGGERED TEST BASIS~~ Frequency. The channel test demonstrates that ~~every~~ each digital automatic actuation logic ~~associated with one of the two safety system trains~~ successfully performs the two-out-of-three logic combinations every 31 days. ~~All automatic actuation logics are thus retested every 62 days.~~ The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the automatic actuation logic. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

~~Automatic actuation logic response time testing is incorporated into the response time testing required by LCO 3.3.5.~~

(continued)

INSERT B3.3-70A

A combination of Required Actions A.1 and A.2 may be used for different components associated with an inoperable digital automatic actuation logic channel.

BASES (continued)

REFERENCES

1. 10 CFR 50.46.
2. ~~BAW 10103A, Rev. 3, July 1977.~~ UFSAR, Chapter 15 (4)
3. 10 CFR 50.36. (2)

B 3.3 INSTRUMENTATION

B 3.3.8 Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)

BASES

BACKGROUND

The EDGs provide a source of emergency power when offsite power is either unavailable or is insufficiently stable to allow safe unit operation. Undervoltage protection will generate a LOPS in the event a loss of voltage or degraded voltage condition occurs in the switchyard. There are two LOPS Functions for each 4.16 kV vital bus.

Three undervoltage relays with [inverse voltage time] characteristics are provided on each 4.16 kV Class 1E instrument bus for the purpose of detecting a sustained undervoltage condition or a loss of bus voltage. The relays are combined in a two-out-of-three logic to generate a LOPS if the voltage is below 75% for a short time or below 90% for a long time. The LOPS initiated ACTIONS are described in FSAR, Section [8.3] (Ref. 1).

Trip Setpoints and Allowable Value

The trip setpoints used in the bistables are based on the analytical limits presented in accident analysis in FSAR, Chapter [14] (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. The actual nominal trip setpoint entered into the bistable is more conservative than that required by the unit specific setpoint calculations. A channel is inoperable if its actuation trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value will assure that limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during anticipated operational occurrences (AOOs); that the consequences of accidents will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or accident; and that the equipment functions as designed.

The undervoltage protection scheme has been designed to protect the unit from spurious trips caused by the offsite power source. This is made possible by the inverse voltage

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Value (continued)

time characteristics of the relays used. A complete loss of offsite power will result in approximately a [1] second delay in LOPS actuation. The EDG starts and is available to accept loads within a 10 second time interval on the Engineered Safety Feature Actuation System (ESFAS) or LOPS. Emergency power is established within the maximum time delay assumed for each event analyzed in the accident analysis (Ref. 2).

With three protection channels in a two-out-of-three trip logic for each division of the 4.16 kV power supply, no single failure will cause or prevent protective system actuation. This arrangement meets IEEE-279-1971 criteria (Ref. 3).

APPLICABLE SAFETY ANALYSES

The EDG LOPS is required for the Engineered Safety Features (ESF) to function in any accident with a loss of offsite power. Its design basis is that of the ESFAS.

Accident analyses credit the loading of the EDG, based on the loss of offsite power, during a loss of coolant accident (LOCA). The actual EDG Start has historically been associated with the ESFAS actuation. The diesel loading has been included in the delay time associated with each safety system component requiring EDG supplied power following a loss of offsite power. The analysis assumes a nonmechanistic EDG loading, which does not explicitly account for each individual component of the loss of power detection and subsequent actions. The total actuation time for the limiting systems, high pressure injection, and low pressure injection is 35 seconds. This delay time includes contributions from the EDG Start, EDG loading, and safety injection system component actuation. The response of the EDG to a loss of power must be demonstrated to fall within this analysis response time when including the contributions of all portions of the delay.

The required channels of LOPS, in conjunction with the ESF systems powered from the EDGs, provide unit protection in the event of any of the analyzed accidents discussed in the accident analysis (Ref. 2), in which a loss of offsite power is assumed.

(continued)

5

BASES

APPLICABLE SAFETY ANALYSES (continued)

The delay times assumed in the safety analysis for the ESF equipment include the 10 second EDG Start delay and, if applicable, the appropriate sequencing delay. The response times for ESFAS actuated equipment in LCO 3.3.5, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation," include the appropriate EDG loading and sequencing delay.

The EDG LOPS channels satisfy Criterion 3 of the NRC Policy Statement.

LCO

The LCO for the LOPS requires that three channels per bus of each LOPS instrumentation Function shall be OPERABLE in MODES 1, 2, 3, and 4 when the LOPS supports safety systems associated with the ESFAS. In MODES 5 and 6, the three channels must be OPERABLE whenever the associated EDG is required to be OPERABLE to ensure that the automatic start of the EDG is available when needed.

Loss of LOPS function could result in the delay of safety systems initiation when required. This could lead to unacceptable consequences during accidents. During the loss of offsite power, which is an AOO, the EDG powers the motor driven emergency feedwater pumps. Failure of these pumps to start would leave only the one turbine driven pump and an increased potential for a loss of decay heat removal through the secondary system.

Only Allowable Values are specified for each Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure that the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within the Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculation. Each Allowable Value specified is more conservative than the analytical limit assumed in the transient and accident analysis to account for instrument uncertainties appropriate to the trip function. These uncertainties are defined in the "[Unit Specific Setpoint Methodology]" (Ref. 4).

(continued)

BASES

LCO (continued)

Degraded Voltage LOPS

Voltage: The minimum Allowable Value includes an allowance for relay coil calibration error and is based on maintaining at least [90%] of rated voltage on the 480 V motor control centers, with a [4.1%] V drop across the [4160/480] V stepdown transformers. The [4.1%] V drop associated with these transformers is the maximum expected due to ESF bus loading, while the MCC contactors are considered to require at least [90%] V for proper operation.

The maximum Allowable Value is not based on equipment operability concerns, but rather avoidance of unnecessary EDG starts due to spurious channel trip.

Time Delay: The response time includes [5 seconds] for undervoltage relay actuation following detection of degraded ES bus voltage, [13 seconds] for the bus trip delay timer, and [3 seconds] for the dead bus timer. Note that the acceptance criteria of [21 seconds] does not account for the setpoint tolerance of [10%] or [± 2.1 seconds].

Loss of Voltage LOPS

Voltage and Response Time: The Allowable Value for the loss of voltage channels is ≥ 0 V. This Allowable Value and the associated channel response time are based on the physical characteristics of the loss of voltage sensing relays. The loss of voltage channels respond to a complete loss of ES bus voltage, providing automatic starting and loading of the associated EDG. However, their response time is not critical to the overall ES equipment response time following an actuation, since the degraded voltage LOPS instrumentation will also respond to the complete loss of voltage, and will do so earlier than the loss of voltage instrumentation. The loss of voltage channel response includes only the time response associated with the undervoltage relays, including the nominal setpoint of [7.8 seconds] and a tolerance of [7%] or [0.55 seconds].

APPLICABILITY

The EDG LOPS actuation Function shall be OPERABLE in MODES 1, 2, 3, and 4 because ESF Functions are designed to provide protection in these MODES. Actuation is also

(continued)

BASES

APPLICABILITY (continued)

required whenever the EDG is required to be OPERABLE by LCO 3.8.2, "AC Sources - Shutdown," so that the EDG can perform its function on a loss of power or degraded power to the vital bus.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the channel is found inoperable, then the function that the channel provides must be declared inoperable and the LCO Condition entered for the particular protection function affected. Since the required channels are specified on a per EDG basis, the Condition may be entered separately for each EDG.

A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each Function.

A.1

If one channel per EDG in one or more Functions is inoperable, it must be tripped within 1 hour. With a channel in trip, the LOPS channels are configured to provide a one-out-of-two logic to initiate a trip of the incoming offsite power. In trip, one additional valid actuation will cause a LOPS signal on the bus. The 1 hour Completion Time is reasonable to evaluate and to take action by correcting a degraded condition in an orderly manner and takes into account the low probability of an event requiring LOPS occurring during this interval.

B.1

Condition B applies when two or more undervoltage or two or more degraded voltage channels on a single bus are inoperable.

Required Action B.1 requires all but one inoperable channel to be restored to OPERABLE status within 1 hour. With two or more channels inoperable, the logic is not capable of providing an automatic EDG LOPS signal for valid loss of voltage or degraded voltage conditions. The 1 hour Completion Time is reasonable to evaluate and to take action by correcting the degraded condition in an orderly manner

(continued)

5

BASES

ACTIONS

B.1 (continued)

and takes into account the low probability of an event requiring LOPS occurring during this interval.

C.1

Condition C applies if the Required Action of Condition A or Condition B and the associated Completion Time is not met.

Required Action C.1 ensures that Required Actions for affected diesel generator inoperabilities are initiated. Depending on unit MODE, the Actions specified in LCO 3.8.1, "AC Sources - Operating," or LCO 3.8.2, are required immediately.

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1

SR 3.3.8.1 is the performance of the CHANNEL CHECK once every 12 hours to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since

(continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with this LCO's required channels.

SR 3.3.8.2

The Note allows channel bypass for testing without defining it as inoperable although during this time period it cannot actuate a diesel start. This allowance is based on the assumption that 4 hours is the average time required to perform channel Surveillance. The 4 hour testing allowance does not significantly reduce the probability that the EDG will start trip when necessary. It is not acceptable to routinely remove channels from service for more than 4 hours to perform required Surveillance testing.

A CHANNEL FUNCTIONAL TEST is performed on each required EDG LOPS channel to ensure the entire channel will perform the intended function. Any setpoint adjustments shall be consistent with the assumptions of the current unit specific setpoint analysis. The Frequency of 31 days is considered reasonable based on the reliability of the components and on operating experience that demonstrates channel failure is rare.

SR 3.3.8.3

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The setpoints and the response to a loss of voltage and a degraded voltage test shall include a single point verification that the trip occurs within the required delay time, as shown in Reference 1. CHANNEL CALIBRATION shall find that measurement setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint analysis in Reference 4.

(continued)

5

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.3 (continued)

The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by the assumption of an 18 month calibration interval in the determination of equipment drift in the setpoint calculation.

REFERENCES

1. FSAR, Section [8.3].
2. FSAR, Chapter [14].
3. IEEE-279-1971, April 1972.
4. [Unit Name], "[Unit Specific Setpoint Methodology]."

B 3.3 INSTRUMENTATION

B 3.3 ~~17~~ 8 2 Post Accident Monitoring (PAM) Instrumentation

BASES

BACKGROUND

The primary purpose of the PAM instrumentation is to display unit variables that provide information required by the control room operators during accident situations. This information provides the necessary support for the operator to take the manual actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Events.

The OPERABILITY of the accident monitoring instrumentation ensures that there is sufficient information available on selected unit parameters to monitor and to assess unit status and behavior following an accident.

The availability of accident monitoring instrumentation is important so that responses to corrective actions can be observed, and so that the need for and magnitude of further actions can be determined. These essential instruments are identified by the ONS specific Regulatory Guide 1.97 analysis ~~[Unit Specific Documents]~~ (Ref. 1) ~~addressing~~ UFSAR, Section 7.5 (Ref. 2), and the NRC's Safety Evaluation Report for the ONS Regulatory Guide 1.97 analysis (Ref. 3) 1 which address the recommendations of Regulatory Guide 1.97 (Ref. ~~2~~ 4) as required by Supplement 1 to NUREG-0737 (Ref. ~~3~~ 5). 2

The instrument channels required to be OPERABLE by this LCO equate to two classes of parameters identified during unit specific implementation of Regulatory Guide 1.97 as Type A and Category ~~I~~ 1 variables. 2

Type A variables are ~~included in this LCO~~ specified 2 because they provide the primary information that permits the control room operator to take specific manually controlled actions that are required when no automatic control is provided and that are required for safety systems to accomplish their safety functions for ~~Design Basis~~ Accidents ~~(DBAs)~~. ~~Because the list of Type A variables widely differs between units, Table 3.3.17-1 in the accompanying LCO contains only those examples of Type A variables that may also be Category I.~~ 2

Category ~~I~~ 1 variables are the key variables deemed risk significant because they are needed to: 2

- Determine whether systems important to safety are performing their intended functions;

(continued)

BASES

BACKGROUND (continued)

- Provide information to the operators that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release; and

- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public and to estimate the magnitude of any impending threat.

These key variables are identified by ~~unit~~ the ONS specific Regulatory Guide 1.97 analysis (Ref. 1). This analysis identifies the unit specific Type A and Category ~~I~~ 1 variables and provides justification for deviating from the NRC proposed list of Category ~~I~~ 1 variables. 2

~~Reviewer's Note: Table 3.3.17-1 provides a list of variables typical of those identified by a unit specific Regulatory Guide 1.97 analysis (Ref. 1). Table 3.3.17-1 in unit specific Technical Specifications shall list all Type A and Category I variables identified by the unit specific Regulatory Guide 1.97 analysis, as amended by the NRC's Safety Evaluation Report (SER).~~ 13

The specific instrument Functions listed in Table 3.3.~~17~~ 8-1 are discussed in the LCO Bases Section. 2

APPLICABLE SAFETY ANALYSES

The PAM instrumentation ensures the availability of information so that the control room operating staff can:

- Perform the diagnosis specified in the emergency operating procedures. These variables are restricted to preplanned actions for the primary success path of 4 accidents ~~DBAs~~ (e.g., loss of coolant accident (LOCA));

- Take the specified, preplanned, manually controlled actions, for which no automatic control is provided, which are required for safety systems to accomplish their safety functions;

- Determine whether systems important to safety are performing their intended functions;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

- Determine the potential for causing a gross breach of the barriers to radioactivity release;
- Determine if a gross breach of a barrier has occurred; and
- Initiate action necessary to protect the public and estimate the magnitude of any impending threat.

2 The ~~unit~~ ONS specific Regulatory Guide 1.97 analysis (Ref. 1) documents the process that identifies Type A and Category ~~1~~ 1 non-Type A variables. 2

PAM instrumentation that meets the definition of Type A in Regulatory Guide 1.97 satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 6). 2 Category ~~1~~ 1 non-type A, instrumentation must be retained in Technical Specifications because it is intended to assist operators in minimizing the consequences of accidents. ~~Therefore,~~ Category 1, non-Type A variables are important for reducing public risk, and therefore, satisfy Criterion 4 of 10 CFR 50.36 (Ref. 6). 2

LCO

LCO 3.3 ~~17~~ 8 requires two OPERABLE channels for all but one Function to ensure no single failure prevents the operators from being presented with the information necessary to determine the status of the unit and to bring the unit to, and maintain it in, a safe condition following that accident. Furthermore, provision of two channels allows a CHANNEL CHECK during the post accident phase to confirm the validity of displayed information. ~~[More than two channels may be required at some units if the Regulatory Guide 1.97 analysis determines that failure of one accident monitoring channel results in information ambiguity (i.e., the redundant displays disagree) that could lead operators to defeat or to fail to accomplish a required safety function.]~~ 13

24 INSERT B 3.3-142A

The exception to the two channel requirement is containment isolation valve position. In this case, the important information is the status of the containment penetrations. 22 The LCO requires one position indicator for each ~~active~~ electrically controlled containment isolation valve. This is sufficient to redundantly verify the isolation status of each isolable penetration either via indicated status of the ~~active~~ valve

(continued)

INSERT B3.3-142A

Where a channel includes more than one control room indication, such as both an indicator and a recorder, the channel is OPERABLE when at least one indication is OPERABLE.

BASES

LCO (continued)

and prior knowledge of the passive valve or via system boundary status. If a normally active containment isolation valve is known to be closed and deactivated, position indication is not needed to determine status. Therefore, the position indication for valves in this state is not required to be OPERABLE.

8 (2)

Each (2) ~~The following list is a discussion~~ of the specified instrument Functions listed in Table 3.3.~~17~~-1 ~~. These discussions are intended as examples of what should be provided for each Function when the unit specific list is prepared.~~ are discussed below: (2)

1. Wide Range Neutron Flux

INSERT B3.3-143A (23)

Wide Range Neutron Flux indication is a Type B, Category 1 variable (2) provided to verify reactor shutdown. ~~[For this unit,~~ the Wide Range Neutron Flux channels consist of ~~the following.]~~

2 ~~3~~. Reactor Coolant System (RCS) Hot ~~and Cold~~ Leg Temperature (5)

RCS Hot ~~and Cold~~ (5) Leg Temperature instrumentation ~~are~~ is a Type B, Category ~~1~~ variable~~s~~ provided for verification of core cooling and long term surveillance. ~~Reactor outlet temperature inputs to the RPS are provided by two fast response resistance elements and associated transmitters in each loop.~~ (4) The channels provide indication over a range of ~~32~~ 50 °F to 700°F.

INSERT B3.3-143B (23)

INSERT from next page

4. RCS Pressure (Wide Range)

RCS Pressure (Wide Range) instrumentation is a Type A, Category 1 variable (2) provided for verification of core cooling and RCS integrity long term surveillance.

Wide range RCS loop pressure is measured by pressure transmitters with a span of 0 psig to 3000 psig. The pressure transmitters are located outside the RB. Redundant monitoring capability is provided by two trains of instrumentation. Control room indications are provided through the inadequate core cooling plasma display. The inadequate core cooling plasma

(continued)

INSERT B3.3-143A

two channels of fission chamber based instrumentation with readout on one recorder. (Note: four channels are available only two are required). The channels provide indication over a range of 1E-8% to 200% RTP.

INSERT B3.3-143B

The two channels provide readout on two indicators. Control room display is through the inadequate core cooling monitoring system.

(23) <Entire Page except as marked>

BASES

LCO

4. RCS Pressure (Wide Range) (continued)

display is the primary indication used by the operator during an accident. Therefore, the accident monitoring specification deals specifically with this portion of the instrument string.

~~In some units~~ RCS Pressure is a Type A, Category 1 variable because the operator uses this indication to monitor the cooldown of the RCS following a steam generator (SG) tube rupture or small break LOCA. Operator actions to maintain a controlled cooldown, such as adjusting SG pressure or level, would use this indication. In addition, high pressure injection (HPI) flow is throttled based on RCS Pressure and subcooled margin. For some small break LOCAs, low pressure injection (LPI) may actuate with ~~system~~ RCS (2) pressure stabilizing above the shutoff head of the LPI pumps. If this condition exists, the operator is instructed to verify HPI flow and then terminate LPI flow prior to exceeding 30 minutes of LPI pump operation against a deadhead pressure. RCS Pressure, in conjunction with LPI flow, is also used to determine if a core flood line break has occurred.

<Move to previous page> (3), 5. Reactor Vessel ~~Water~~ Head Level and RCS Hot Leg Level

Reactor Vessel ~~Water~~ Level instrumentation is a Type B, Category 1 variable provided for verification and long term surveillance of core cooling. The reactor vessel level monitoring system provides ~~a direct measurement~~ an indication of the ~~collapsed~~ liquid level ~~above the fuel alignment plate. The collapsed level represents the amount of liquid mass that is in the reactor vessel above the core. Measurement of the collapsed water level is selected because it is a direct indication of the water inventory.~~

~~The collapsed level is obtained over the same temperature and pressure range as the saturation measurements, thereby encompassing all operating and accident conditions where it must function. Also, it functions during the recovery interval. Therefore, it is designed to survive the high steam temperature that may occur during the preceding core recovery interval.~~

INSERT B 3.3-144A

(continued)

INSERT B3.3-144A

from the top of the Hot Leg on each steam generator to the bottom of the Hot Leg as it exits the vessel and from the top of the reactor vessel head to the bottom of the Hot Leg as it exits the vessel. Compensation is provided for impulse line temperature variations.

BASES

and RCS Hot Leg Level

LCO 3, 5. Reactor Vessel ~~Water~~ Head Level (continued)

The level range extends from the top of the vessel down to the top of the fuel alignment plate. The response time is short enough to track the level during small break LOCA events. The resolution is sufficient to show the initial level drop, the key locations near the hot leg elevation, and the lowest levels just above the alignment plate. This provides the operator with adequate indication to track the progression of the accident and to detect the consequences of its mitigating actions or the functionality of automatic equipment.

~~[For this unit,]~~ The Reactor Vessel Water Level channels consist of ~~the following:~~ 1

INSERT B3.3-145A

6. Containment Sump Water Level (Wide Range)

a Type B, Category I variable

Containment Sump Water Level (Wide Range) instrumentation is provided for verification and long term surveillance of RCS integrity. ~~[For this unit,]~~ The Containment Sump Water Level instrumentation consists of ~~the following:]~~ 1

INSERT B3.3-145B

7. Containment Pressure (Wide Range)

Containment Pressure (Wide Range) instrumentation is provided for verification of RCS and containment OPERABILITY. ~~[For this unit,]~~ Containment Pressure instrumentation consists of ~~the following:]~~ 1

INSERT B3.3-145C

8. Containment Isolation Valve Position

2 Containment isolation valve (~~P~~CIV) position is provided for verification of ~~containment integrity~~ electrically controlled containment isolation valve position. In the case of ~~P~~CIV position, the important information is the isolation status of the containment penetration. The LCO requires one channel of valve position indication in the control room to be OPERABLE for each ~~active~~ ~~P~~CIV in a containment penetration flow path, i.e., two total channels of ~~P~~CIV position indication for a penetration flow path with two ~~active~~ electrically controlled valves. For containment

(continued)

INSERT B3.3-145A

two Reactor Vessel Head Level channels that provide readout on two indicators (RC-LT0125 and RC-LT0126) with one channel recorded in the control room and two RCS Hot Leg Level channels that provide readout on two indicators (RC-LT0123 and RC-LT0124) with one channel recorded in the control room.

INSERT B3.3-145B

two channels with readout on two indicators (LT-90 and LT-91) and one recorder. The indicated range is 0 to 15 feet.

INSERT B3.3-145C

two channels with readout on two indicators (PT-230 and PT-231) and one channel recorded. The indicated range is -5.0 psig to 175 psig.

23 <Except as marked>

electrically controlled 22

BASES

LCO

8. Containment Isolation Valve Position (continued)

penetrations with only one ~~active~~ PCIV having control room indication, Note (b) requires a single channel of valve position indication to be OPERABLE. This is sufficient to redundantly verify the isolation status of each isolable penetration via indicated status of the ~~active~~ valve, as applicable, and prior knowledge of passive valve or system boundary status. ~~If a penetration flow path is isolated, position indication for the PCIV(s) in the associated penetration flow path is not needed to determine status.~~ Therefore, the position indication for valves in an isolated penetration flow path is not required to be OPERABLE. ~~[For this plant,~~ the PCIV position PAM instrumentation consists of ~~the following:]~~ 1

22 electrically controlled

INSERT B 3.3-146A

INSERT B3.3-146B

INSERT B 3.3-146C

9. Containment Area Radiation (High Range)

a Type C, Category 1 variable

Containment Area Radiation (High Range) instrumentation is ~~provided~~ to monitor the potential for significant radiation releases and to provide release assessment for use by operators in determining the need to invoke site emergency plans. ~~[For this unit,~~ the Containment Area Radiation instrumentation consists of ~~the following:]~~ 1

INSERT B3.3-146D

10. Containment Hydrogen Concentration

a Type A, Category 1 variable

Containment Hydrogen Concentration instrumentation is provided to detect high hydrogen concentration conditions that represent a potential for containment breach. This variable is also important in verifying the adequacy of mitigating actions. ~~[For this unit,~~ the Containment Hydrogen Concentration instrumentation consists of ~~the following:]~~ 1

INSERT B3.3-146E

11. Pressurizer Level

Pressurizer Level instrumentation is used, to determine whether to terminate safety injection (SI), if still in progress, or to reinitiate SI if it has been stopped. Knowledge of pressurizer water level is also

in combination with other system parameters

(continued)

INSERT B3.3-146A

As indicated by Note (a) to the Required Channels, if a penetration flow path is isolated by at least one closed and deactivated automatic valve, closed manual valve, blind flange, or check valve with flow through the valve secured, position indication for the CIV(s) in the associated penetration flow path is not needed to determine status.

INSERT B3.3-146B

Note (c) to the Required Channels indicates that position indication requirements apply only to CIVs that are electrically controlled.

INSERT B3.3-146C

limit switches that operate both Closed-Not Closed and Open-Not Open control switch indication via indicating lights in the control room.

INSERT B3.3-146D

two channels (RIA 57 and 58) with readout on two indicators and one channel recorded. The indicated range is 1 to 10^7 R/hr.

INSERT B3.3-146E

two channels with readout on two indicators and one channel recorded. The indicated range is 0 to 10% hydrogen concentration.

(23) <Except as marked>

BASES

LCO 11. Pressurizer Level (continued)

used to verify the unit conditions necessary to establish natural circulation in the RCS and to verify that the unit is maintained in a safe shutdown condition. ~~[For this unit,~~ the Pressurizer Level instrumentation consists of ~~the following:]~~ (1)

INSERT B 3.3-147A

12. Steam Generator Water Level

a Type A, Category 1 variable

Steam Generator Water Level instrumentation is provided to monitor operation of decay heat removal via the SG. The indication of SG level is the extended startup range level instrumentation, covering a span of ~~6~~ 0 inches to ~~394~~ 388 inches above the lower tubesheet. ~~The measured differential pressure is displayed in inches of water at 68°F. Temperature compensation for this indication is performed manually by the operator. Redundant monitoring capability is provided by two trains of instrumentation. The uncompensated level signal is input to the unit computer, a control room indicator, and the Emergency Feedwater (EFW) Control System.~~

~~SG level indication is used by the operator to manually raise and control SG level to establish boiler condenser heat transfer. Operator action is initiated on a loss of subcooled margin. Feedwater flow is increased until the indicated extended startup range level reaches the boiler condenser setpoint.~~

INSERT B 3.3-147B

15 ~~13.~~ ~~Condensate Storage Tank (CST)~~ Upper Surge Tank (UST) Level

INSERT B 3.3-147C

~~CST~~ level instrumentation is a Type A, Category 1 variable provided to ensure ~~a~~ water supply for EFW. ~~The CST provides the assured, safety grade water supply for the EFW System. The CST consists of two identical tanks connected by a common outlet header. Inventory is monitored by a 0 inch to 144 inch level indication for each tank. CST Level is displayed on a control room indicator, strip chart recorder, and unit computer. In addition, a control room annunciator alarms on low level.~~

INSERT B 3.3-147D

(continued)

INSERT B3.3-147A

three channels (two for Train A and one for Train B) on the computer and one channel recorded (selected among the three channels). The indicated range is 0 to 400 inches (11% to 84% level as a percentage of volume).

INSERT B3.3-147B

The operator relies upon SG level information following an accident (e.g., main steam line break, steam generator tube rupture) to isolate the affected SG to confirm adequate heat sinks for transients and accidents.

The extended startup range Steam Generator Level instrumentation consists of four indicators (2 per steam generator). The channels also display on the computer and and one channel provides input to a recorder.

INSERT B3.3-147C

13. Steam Generator Pressure

Steam Generator Pressure instrumentation is a Type A, Category 1 variable provided to support operator diagnosis of a main steam line break or SG tube rupture accident to identify and isolate the affected SG. In addition, SG pressure is a key parameter used by the operator to evaluate primary-to-secondary heat transfer.

Steam generator pressure measurement is provided by two pressure transmitters per SG. Each instrument channel inputs to the ICCM cabinet that provide safety inputs to two indicators located on the main control board in the control room. One channel per SG also provides input to a recorder located in the control room.

14. Borated Water Storage Tank (BWST) Level

BWST Level instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements, i.e., to determine when to initiate the switch over of the core cooling pump suction from the BWST to sump recirculation. BWST level measurement is provided by two channels with readout on two indicators and one channel recorded. The channels provide level indication over a range of 0 to 50 feet (13% to 100% of volume).

INSERT B3.3-147D

EFW draws condensate grade suction from the USTs and the Condenser Hotwell.

Two Category 1 instrumentation channels are provided for monitoring UST level. These instrument channels are inputs to corresponding train A and B Inadequate Core Cooling Monitoring (ICCM) system cabinets. The ICCM Train A cabinet provides UST level input to a dedicated qualified recorder and to a qualified indicator, both located in the Control Room. The ICCM Train B cabinet also provides an input to a qualified indicator located in the Control Room. The range of UST level indication is 0 to 12 feet.

BASES

LCO

15 ~~13.~~ Upper Surge Tank (UST) ~~Condensate Storage Tank (CST)~~ Level (continued)

UST ~~CST~~ Level is the primary indication used by the operator to identify loss of ~~CST~~ UST volume. ~~and~~ The operator can then decide to replenish the ~~CST~~ UST or align suction to the EFW pumps from the hotwell.

16 ~~14.~~ Core Exit Temperature

Core Exit Temperature is a Type A, Category 1 variable provided for verification and long term surveillance of core cooling. INSERT B 3.3-148A ~~An evaluation was made of the minimum number of valid core exit thermocouples (CETs) necessary for inadequate core cooling detection. The evaluation determined the reduced complement of CETs necessary to detect initial core recovery and to trend the ensuing core heatup. The evaluations account for core nonuniformities and cold leg injection. Based on these evaluations, adequate or inadequate core cooling detection is ensured with two sets of five valid CETs.~~

Table 3.3.8-1 Note (d) indicates that ~~T~~the subcooling margin monitor takes the average of the five highest CETs for each of the ICCM trains. Two channels ensure that a single failure will not disable the ability to determine the representative core exit temperature.

INSERT B 3.3-148B

21 ~~15.~~ Emergency Feedwater Flow

EFW Flow instrumentation is a Type D, Category 1 variable provided to monitor operation of ~~decay~~ RCS heat removal via the SGs. ~~The~~ Two channels provide indication of EFW Flow to each SG ~~is determined from a differential pressure measurement calibrated to a span of 0 gpm to 1200 gpm~~ over a range of approximately 100 to 1200 gpm. Redundant monitoring capability is provided by two independent ~~trains~~ channels of instrumentation for each SG. Each ~~differential~~ the pressure transmitter provides an input to a control room indicator. ~~and the unit computer.~~ One channel also provides input to a recorder.

EFW Flow is the primary indication used by the operator to ~~determine the need to throttle flow during an SLB accident to prevent the EFW pumps from operating in runout conditions. EFW Flow is also used by the operator to~~ verify that the EFW System is

(continued)

INSERT B 3.3-148A

The operator relies on this information following a LOCA to secure HPI and throttle LPI, following a SBLOCA to throttle HPI and begin forced HPI cooling if needed, and following a MSLB and SG Tube Rupture to throttle HPI and isolate the affected SG.

There are a total of 52 Core Exit Thermocouples (CETs) per Oconee Unit. Twenty-four (12 per train) meet seismic and environmental qualification requirements (Category 1). The unit computer is the primary display for all 52 CETs. The CETs are distributed to provide monitoring of four or more in each quadrant for each train. The ICCM plasma displays (1 per train) located in the Control Room serve as safety related backup displays for the twenty-four Category 1 CETs. The range of the readouts is 50°F to 2300°F.

The ICCM CET function uses inputs from twelve incore thermocouples per train to calculate and display temperatures of the reactor coolant as it exits the core and to provide indication of thermal conditions across the core at the core exit. Each of the twelve qualified thermocouples per train is displayed on a spatially oriented core map on the plasma display. Trending of CET temperature is available continuously on the plasma display. The average of the five hottest CETs is trendable for the past forty minutes.

INSERT B 3.3-148B

17. Subcooling Monitor

The Subcooling Monitor is a Type A, Category 1 variable provided for verification and long term surveillance of core cooling. This variable is a computer calculated value using various inputs from the Primary System.

Two channels of indication are provided. One channel monitors RCS Loop A and the Core Saturation margin while another separate channel monitors RCS Loop B and the Core Saturation margin. The indication readouts are located in the control room. This variable also inputs to the unit computer through isolation buffers and is available for trend recording upon operator demand. The range of the readouts is 200°F subcooled to 50°F superheat. The control room display is through the ICCM plasma display unit.

A backup method for determining subcooling margin ensures the capability to accurately monitor RCS subcooling margin (Refer to Specification 5.5.17).

INSERT B 3.3-148B (continued)

18. HPI System Flow

HPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for short term cooling requirements, to prevent HPI pump runout and inadequate NPSH, and to indicate the need for flow cross connect. HPI flow is throttled based on RCS pressure, subcooled margin, and pressurizer level. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two HPI trains. The channels provide flow indication over a range of 0 to 750 gpm.

19. LPI System Flow

LPI System Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements, to prevent LPI pump runout and for flow balance. The indication is also used to identify an LPI pump operating at system pressures above its shutoff head. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two LPI trains. The LPI channels provide flow indication over a range of 0 to 6000 gpm.

20. Reactor Building Spray Flow

Reactor Building Spray Flow instrumentation is a Type A, Category 1 variable provided to support action for long term cooling requirements and iodine removal and to prevent Reactor Building Spray and LPI pump runout. Flow measurement is provided by one channel per train with readout on an indicator and recorder. There are two RBS trains. The channels provide flow indication over a range from 0 to 2000 gpm.

BASES

LCO

(21) ~~15~~ Emergency Feedwater Flow (continued)

delivering the correct flow to each SG. However, the primary indication used by the operator to ensure an adequate inventory is SG level.

RCS pressure is used by the operator to monitor the cooldown of the RCS following an SG tube rupture or small break LOCA. In addition, HPI flow is throttled based on RCS pressure and subcooled margin. The indication is also used to identify an LPI pump operating at system pressures above its shutoff head. If this condition exists, the operator is instructed to verify this condition exists, to verify HPI flow, and to terminate LPI flow prior to exceeding 30 minutes of LPI pump operation against a deadhead pressure. RCS pressure, in conjunction with LPI flow, is also used to determine if a core flood line break has occurred. (23)

APPLICABILITY

The PAM instrumentation LCO is applicable in MODES 1, 2, and 3. These variables are related to the diagnosis and preplanned actions required to mitigate accidents and transients (4) ~~DBAs~~. The applicable ~~DBAs~~ accidents and transients (4) are assumed to occur in MODES 1, 2, and 3. In MODES 4, 5, and 6, unit conditions are such that the likelihood of an event occurring that would require PAM instrumentation is low; therefore, the PAM instrumentation is not required to be OPERABLE in these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to the ACTIONS to exclude the MODE change restriction of LCO 3.0.4. This exception allows entry into an applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require a unit shutdown. This exception is acceptable due to the passive function of the instruments, the operator's ability to respond to an accident utilizing alternate instruments and methods, and the low probability of an event requiring these instruments.

Note ~~Two~~ 2 is added to the ACTIONS to clarify the application (2) of Completion Time rules. The Conditions of this Specification may be entered independently for each

(continued)

BASES

ACTIONS (continued)

Function listed in Table 3.3.~~17~~ 8-1. (2) The Completion Time(s) of the inoperable channels of a Function ~~will be~~ are (2) tracked separately for each Function starting from the time the Condition ~~was~~ is (2) entered for that Function.

A.1

When one or more Functions have one required channel inoperable, the inoperable channel must be restored to OPERABLE status within 30 days. The 30 day Completion Time is based on operating experience. This takes into account the remaining OPERABLE channel (or, in the case of a Function that ~~has only one~~ required channel, other non-Regulatory Guide 1.97 instrument channels to monitor the Function), the passive nature of the instrument (no critical automatic action is assumed to occur from these instruments), and the low probability of an event requiring PAM instrumentation during this interval. (2)

(5) INSERT B 3.3-150 A (2)

B.1

Required Action B.1 specifies (6) initiation of action described in Specification 5.6.~~8~~, that requires a written report to be submitted to the NRC. This report discusses the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. This action is appropriate in lieu of a shutdown requirement since alternative actions are identified before loss of functional capability and given the likelihood of unit conditions that would require information provided by this instrumentation. The Completion Time of "Immediately" for Required Action B.1 ensures the requirements of Specification 5.6.~~8~~ 6 are initiated. (2)

C.1

When one or more Functions have two required channels inoperable (i.e., two channels inoperable in the same Function), one channel in the Function should be restored to OPERABLE status within 7 days. This Condition does not apply to the hydrogen monitor channels. The Completion Time of 7 days is based on the relatively low probability

(continued)

INSERT B 3.3-150A

Condition A is modified by a Note indicating this Condition is not applicable to PAM Functions 14, 18, 19, and 20.

BASES

ACTIONS

C.1 (continued)

of an event requiring PAM instrumentation action operation and the availability of alternative means to obtain the required information. Continuous operation with two required channels inoperable in a Function is not acceptable because the alternate indications may not fully meet all performance of qualification requirements applied to the PAM instrumentation. Therefore, requiring restoration of one inoperable channel of the Function limits the risk that the PAM Function will be in a degraded condition should an accident occur.

5 INSERT B3.3-151A

D.1

When two required hydrogen monitor channels are inoperable, Required Action D.1 requires one channel to be restored to OPERABLE status. This action restores the monitoring capability of the hydrogen monitor. The 72 hour Completion Time is based on the relatively low probability of an event requiring hydrogen monitoring ~~and the availability of alternative means to obtain the required information~~. Continuous operation with two required channels inoperable 4 is not acceptable because alternate indications are not available.

5 INSERT B 3.3-151B

G ~~E~~.1 5

Required Action G ~~E~~.1 directs entry 8 into the appropriate Condition referenced in Table 3.3.~~17~~-1. The applicable Condition referenced in the Table is Function dependent. Each time an inoperable channel has not met ~~any~~ the Required Action and associated Completion Time of Condition C ~~or~~ D, E as applicable, ~~and the associated Completion Time has expired~~, Condition ~~E~~ G is entered for that channel and provides for transfer to the appropriate subsequent Condition. 5 2

H ~~F~~.1 and H.2 5

If the Required Action or E and associated Completion 8 2 Time of Conditions C, ~~or~~ D are not met and Table 3.3.~~17~~-1 directs entry into Condition ~~F~~ H 5, the unit must be brought to a MODE in which the requirements of this LCO do not apply. To

(continued)

INSERT B 3.3-151A

Condition C is modified by a Note indicating this Condition is not applicable to PAM Functions 10, 14, 18, 19, and 20.

INSERT B 3.3-151B

Condition D is modified by a Note indicating this Condition is only applicable to PAM Function 10.

E.1

When one required BWST water level channel is inoperable, Required Action E.1 requires the channel to be restored to OPERABLE status. The 24 hour Completion Time is based on the relatively low probability of an event requiring BWST water and the availability of the remaining BWST water level channel. Continuous operation with one of the two required channels inoperable is not acceptable because alternate indications are not available. This indication is crucial in determining when the water source for ECCS should be swapped from the BWST to the reactor building sump.

Condition E is modified by a Note indicating this Condition is only applicable to PAM Function 14.

F.1

When a flow instrument channel is inoperable, Required Action F.1 requires the affected HPI, LPI, or RBS train to be declared inoperable and the requirements of LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 apply. The required Completion Time for declaring the train(s) inoperable is immediately. Therefore, LCO 3.5.2, LCO 3.5.3, or LCO 3.6.5 is entered immediately, and the Required Actions in the LCOs apply without delay. This action is necessary since there is no alternate flow indication available and these flow indications are key in ensuring each train is capable of performing its function following an accident. HPI, LPI, and RBS train OPERABILITY assumes that the associated PAM flow instrument is OPERABLE because this indication is used to throttle flow during an accident and assure runout limits are not exceeded or to ensure the associated pumps do not exceed NPSH requirements.

Condition F is modified by a Note indicating this Condition is only applicable to PAM Functions 18, 19, and 20.

BASES

ACTIONS

~~F~~ H.1 and H.2 (continued) 5

achieve this status, the unit must be brought to at least MODE 3 within ~~6~~ 12 hours and MODE 4 within ~~12~~ 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. 5

~~G~~ I.1 5

2 INSERT B 3.3-152A

At this unit, alternative means of monitoring Containment Area Radiation ~~have been developed and tested~~. These alternative means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allowed time. 2

~~If these alternative means are used,~~ the Required Action is not to shut the unit down, but rather to follow the directions of Specification 5.6.~~8~~, in the Administrative Controls section of the Technical Specifications. The report provided to the NRC should discuss the alternative means used, describe the degree to which the alternative means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels. 6 2

In the case of reactor vessel level, Reference 4 determined that the appropriate Required Action was not to shut the unit down, but rather to follow the directions of Specification 5.6.8. 2

INSERT B 3.3-152B

[~~At this unit,~~ the alternative monitoring provisions consist of the following:] 1

SURVEILLANCE REQUIREMENTS

As noted at the beginning of the SRs, the SRs apply to each PAM instrumentation Function in Table 3.3.~~17~~ 8-1, except where indicated. 2

(continued)

INSERT B 3.3-152A

If the Required Action and associated Completion Time of Condition C, D or E are not met and Table 3.3.8-1 directs entry into Condition I, alternate means of monitoring the parameter should be applied and

INSERT B 3.3-152B

Both the RCS Hot Leg Level and the Reactor Vessel Level are methods of monitoring for inadequate core cooling capability. The subcooled margin monitors (SMM), and core-exit thermocouples (CET) provide an alternate means of monitoring for this purpose. The function of the ICC instrumentation is to increase the ability of the unit operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory.

The alternate means of monitoring the Reactor Building Area Radiation (High Range) consist of a combination of installed area radiation monitors and portable instrumentation.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.~~17~~ 8 .1 - 2

Performance of the CHANNEL CHECK once every 31 days for each required instrumentation channel that is normally energized ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel with a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are ~~determined by the unit staff~~ 2 based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Offscale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale. 2

The Frequency is based on ~~unit~~ 2 operating experience that demonstrates channel failure is rare. The CHANNEL CHECK supplements less formal but more frequent checks of channels during normal operational use of the displays associated with this LCO's required channels.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3 ~~17~~ 8 .2 and SR 3.3.8.3 5 2

~~A CHANNEL CALIBRATION is performed every [18] months or approximately at every refueling.~~ 5 A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. This test verifies the channel responds to measured parameters within the necessary range and accuracy. A 2

~~A~~ Note to SR 3.3.8.3 2 clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices, with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration.

TSTF-019, Rev. 1

INSERT B3.3-154A

For the Containment Area Radiation instrumentation, a CHANNEL CALIBRATION may consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr, and a one point calibration check of the detector below 10 R/hr with a gamma source.

5 INSERT B3.3-154B

The Frequency of each SR 2 ~~is based on operating experience and consistency with the typical industry refueling cycle~~ 2 and is justified by the assumption of ~~an [18] month~~ the specified 2 calibration interval in the determination of the magnitude of equipment drift.

REFERENCES

1. ~~[Unit Specific Documents (e.g., FSAR, NRC Regulatory Guide 1.97 SER letter).]~~ 1

INSERT B 3.3-154C

~~2~~ 4. Regulatory Guide 1.97.

INSERT B 3.3-154D

~~3~~ 5. NUREG-0737, ~~1979~~. INSERT B3.3-154E

~~4. 32-1177256-00, "Technical Basis for Reactor Vessel Level Indication System (RVLIS) Action Statement," April 10, 1990.~~ 2

6. 10 CFR 50.36

TSTF-019, Rev. 1

INSERT B 3.3-154A

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature detectors (RTD)sensors or Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

2

~~Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the Core Exit thermocouple sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.~~

INSERT B 3.3-154B

SR 3.3.8.2 is modified by a Note indicating that it is applicable only to Functions 7 and 10. SR 3.3.8.3 is modified by Note 2 indicating that it is not applicable to Functions 7 and 10.

INSERT B 3.3-154C

1. Duke Power Company letter from Hal B. Tucker to Harold M. Denton (NRC) dated September 28, 1984.
2. UFSAR, Section 7.5.
3. NRC Letter from Helen N. Pastis to H. B. Tucker, "Emergency Response Capability - Conformance to Regulatory Guide 1.97," dated March 15, 1988.

INSERT B 3.3-154D

, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident," Revision 3, May 1983.

INSERT B 3.3-154E

"Clarification of TMI Action Plan Requirements," 1980.

(4) <Except as marked>

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality at lower power levels than can be seen on the ~~intermediate~~ wide range neutron flux instrumentation. These channels also provide the operator with a flux indication that reveals changes in reactivity and helps to verify that SDM is being maintained.

The source range instrumentation has ~~two~~ four redundant count rate channels originating in ~~two high sensitivity~~ ~~proportional counters~~ fission chambers. ~~Two~~ Four source range detectors are externally located ~~on opposite sides of~~ symmetrically around the core ~~180°~~. These channels are used over a counting range of 0.1 cps to 1E~~6~~5 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5(1) decades to +5(7) decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel.

~~The proportional counters of the source range channels are BF_3 chambers. The detector high voltage is automatically turned off when the flux level is approximately one decade above the useful operating range. Conversely, the high voltage is turned on automatically when the flux level returns to within approximately one decade of the detectors' maximum useful range. High voltage will be turned off automatically when the flux level is above 1E-10 amp in both intermediate range channels, or 10% power in power range channels.~~ (15)

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels ~~are~~ They are (2) necessary to monitor core reactivity changes. ~~It is~~ the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. ~~It~~ They (2) also trigger~~s~~ operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The source range neutron flux channels satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 1). — 2

5 4E-4% RTP

LCO

Two source range neutron flux channels shall be OPERABLE ~~whenever the control rods are capable of being withdrawn~~ to provide the operator with redundant source range neutron instrumentation. The source range instrumentation ~~is the~~ provides primary power indication at low power levels < ~~1E-10 amp~~ on ~~intermediate~~ wide range instrumentation and must remain OPERABLE for the operator to continue increasing power.

2 — 2 — 4

~~A Note has been added allowing detector high voltage to be de-energized above 1E-10 amp on the intermediate range channels. Above this point, the source range instrumentation is no longer the primary power indicator. As such, the high voltage to the source range detectors may be de-energized.~~ — 5 — 15

APPLICABILITY

Two source range neutron flux channels shall be OPERABLE in MODE 2 to provide redundant indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the ~~intermediate~~ wide range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1.

4 — wide

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring ~~changes in SDM~~ neutron flux and to provide an early indication of reactivity changes. — 2

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2, "Nuclear Instrumentation." — 2

ACTIONS

A.1

The Required Action for one required channel of the source range neutron flux indication inoperable with THERMAL POWER ≤ ~~1E-10 amp~~ 4E-4% RTP on the ~~intermediate~~ wide range neutron flux — 5 — 4 — 5

(continued)

BASES

ACTIONS

A.1 (continued)

instrumentation is to delay increasing reactor power until the channel is repaired and restored to OPERABLE status. This limits power increases in the range where the operators rely solely on the source range instrumentation for power indication. The Completion Time ensures the source range is available prior to further power increases. Furthermore, it ensures that power remains below the point where the ~~intermediate~~ wide (4) range channels provide primary protection ~~until both source range channels are available to support the overlap verification required by SR 3.3.9.4~~.

B.1, B.2, B.3, and B.4

With both required (5) source range neutron flux channels inoperable with THERMAL POWER ≤ ~~1E-10 amp~~ 4E-4% RTP (5) on the ~~intermediate~~ wide (4) range neutron flux instrumentation, the operators must ~~place the reactor in the next lowest condition for which source range instrumentation is not required~~ take actions to limit the possibilities for adding positive reactivity. (16) This is done by immediately suspending positive reactivity additions, initiating action to insert all CONTROL RODS, and opening the CONTROL ROD drive trip breakers within 1 hour. TSTF-009, Rev. 1 Periodic SDM verification ~~of ≥ 1% Δk/k~~ (2) is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than control rod withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in this condition, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required Action B.1, Required Action B.2, and Required Action B.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action B.3 and Required Action B.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification ~~ensures~~ provides reasonable assurance (2) that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged.

C.1

With reactor power > ~~1E-10 amp~~ 4E-4% RTP (5) in MODE 2, 3, 4, or 5 on the ~~intermediate~~ wide (4) range neutron flux instrumentation, continued

(continued)

BASES

ACTIONS

C.1 (continued)

required 5

operation is allowed with one or more source range neutron flux channels inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the channel(s) to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until channels are restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

2

Agreement criteria are determined ~~by the unit staff~~, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the ~~transmitter or the~~ signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. ~~Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.~~ 17

equivalent to

The Frequency, ~~about once~~ every shift, is based on operating experience that demonstrates channel failure is rare. Since 2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially ~~more frequent~~, checks of channel OPERABILITY during ~~normal operational use of the~~ displays associated with the LCO's required channels. When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant source ~~range is~~ may not be available for comparison. CHANNEL CHECK may ~~still be~~ be performed via comparison with ~~intermediate~~ wide range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status. (2) (4)

SR 3.3.9.2

For source range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels from the preamplifier input to the indicators. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. The detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of ~~[18]~~ 24 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an ~~[18]~~ 24 month interval, such that the instrument is not adversely affected by drift. (1)

SR 3.3.9.3

SR 3.3.9.3 is the verification of one decade of overlap with the intermediate range neutron flux instrumentation prior to (5)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.3 (continued)

~~source range count rate exceeding 10^5 cps if not performed within 7 days prior to reactor startup. This ensures a continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the unit in a safe, subcritical condition until the verification can be made. The test may be omitted if performed within the previous 7 days based on operating experience, which shows that source range and intermediate range instrument overlap does not change appreciably within this test interval.~~ 5

REFERENCES

~~None.~~ 1. 10 CFR 50.36 2

4 <Except as marked>

B 3.3 INSTRUMENTATION

B 3.3.10 wide ~~Intermediate~~ Range Neutron Flux

INSERT B 3.3-86A

BASES

BACKGROUND

The wide ~~intermediate~~ range neutron flux channels provide the operator with an indication of reactor power ~~at higher power levels than the source range instrumentation and lower power levels than the power range instrumentation~~.

The wide ~~intermediate~~ range four ~~instrumentation~~ has four ~~two~~ log N channels originating in ~~two~~ electrically identical fission ~~gamma compensated ion~~ chambers. Each channel provides ten ~~eight~~ decades of flux level information in terms of the log of ion chamber count rate and startup rate ~~current from 1E-10 amp to 1E-2 amp~~. ~~The channels also~~ The startup rate, which measures the rate of change of the neutron flux level, ~~which~~ is displayed for the operator in a range ~~terms of startup rate~~ from -0.5 decades to +5 ~~7~~ decades per minute. A high startup rate of +2 ~~3~~ decades per minute in either channel will initiate a control rod withdrawal inhibit.

~~The intermediate range compensated ion chambers are of the electrically adjustable gamma compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.~~

APPLICABLE SAFETY ANALYSES

wide ~~Intermediate~~ range neutron flux channels are necessary to monitor core reactivity changes and are the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions.

The wide ~~intermediate~~ range neutron flux channels satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 1). 2

LCO

Two wide ~~intermediate~~ range neutron flux instrumentation channels shall be OPERABLE to provide the operator with redundant neutron flux indication. These enable operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling

(continued)

INSERT B3.3-86A

from 1E-8 to 200% of RTP and fully overlap the source range and power range channels providing continuity of information needed during startup.

4

5 <except as marked>

BASES

LCO (continued)

neutron flux transients that could result in reactor trip during power escalation.

APPLICABILITY

The ~~intermediate~~ wide range neutron flux channels shall be OPERABLE in MODE 2 and in MODES 3, 4, and 5 ~~when~~ with any CONTROL ROD drive (CRD) trip breaker ~~is~~ in the closed position and the CRD System is capable of rod withdrawal. 5 2

The ~~intermediate~~ wide range instrumentation is designed to detect power changes during initial criticality and power escalation when the power range and source range instrumentation cannot provide reliable indications. Since those conditions can exist in all of these MODES, the ~~intermediate~~ wide range instrumentation, or propagate from, must be OPERABLE. 2

4

ACTIONS

A.1

If one required ~~intermediate~~ wide range channel becomes inoperable ~~when the channels indicate 1E-10 amp~~, the unit is exposed to the possibility that a single failure will disable all neutron monitoring instrumentation. To avoid this, the inoperable channel must be repaired or power must be reduced to the point where source range channels can provide neutron flux indication. Completion of Required Action A.1 places the unit in this state, and LCO 3.3.9, "Source Range Neutron Flux," requires OPERABILITY of two source range ~~detectors~~ channels once this state is reached. If the one channel failure occurs when indicated power is < 4E-4% RTP ~~1E-10 amp~~, the Required Action prohibits increases in power above the source range capability. 4 2 2

The 2 hour Completion Time allows controlled reduction of power into the source range and is based on unit operating experience that demonstrates the improbability of the second ~~intermediate~~ wide range channel failing during the allowed interval. 4

B.1 and B.2

With two required ~~intermediate~~ wide range neutron flux channels inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the 4

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

reactor in the next lowest condition for which the (4) wide ~~intermediate~~ range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

(2)

Agreement criteria are ~~determined by the unit staff~~ based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the ~~transmitter or the~~ signal processing equipment has (17) drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

equivalent to (2)

The Frequency, ~~about once~~ every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1 (continued)

potentially (2)

failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

When operating in Required Action A.1, CHANNEL CHECK is still required. However, in this condition, a redundant (4) wide ~~intermediate~~ range ~~is not~~ (2) may not be available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE ~~intermediate~~ wide (4) range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

For ~~intermediate~~ wide range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an ~~[~~18~~]~~ month interval such that the instrument is not adversely affected by drift. (1)

SR 3.3.10.3

SR 3.3.10.3 is the verification ~~within 7 days prior to reactor startup~~ once each reactor startup of one decade of overlap with the ~~power~~ source (5) range neutron flux instrumentation ~~prior to intermediate range indication exceeding 1E-6 amp~~. This ensures a

(5) INSERT B3.3-89A

(continued)

INSERT B3.3-89A

The wide range detector should be on scale and indicating ≥ 1E-8% of RTP when the source range detector is indicating ≤ 10^4 counts per second in order for the wide range detector to indicate a one decade change prior to the source range detector going off scale.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.3 (continued)

continuous source of power indication during the approach to criticality. ~~Failure to perform this Surveillance leaves the unit in a condition where the intermediate range channels provide adequate protection until the verification can be made.~~ 5

The test may be omitted if performed within the previous 7 days based on operating experience, which shows that the source range and 4 wide ~~intermediate~~ range instrument overlap does not change appreciably within this test interval.

REFERENCES

~~None.~~ 10 CFR 50.36. 2

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(4) <Entire Page>

Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation

B 3.3 INSTRUMENTATION

B 3.3.11 Emergency Feedwater Initiation and Control (EFIC) Instrumentation

BASES

BACKGROUND

The EFIC System instrumentation is designed to provide ~~safety grade means of controlling the secondary system~~ as a heat sink for core decay heat removal. To ensure the secondary system remains a heat sink, the EFIC System takes action to initiate emergency feedwater (EFW) when the primary source of feedwater is lost and to isolate functional components from hydraulic faults within the secondary system. These actions ensure that a source of cooling water is available to be fed to a once through steam generator (OTSG) that has a controlled steam pressure, thereby fixing the heat sink temperature at the saturation temperature of the secondary system. The EFIC Functions that are supported and the parameters that are needed for each of these Functions are described next.

INSERT B 3.3-91/A

The EFIC instrumentation contains devices and circuitry that generate the following signals when monitored variables reach levels that are indicative of conditions requiring protective actions.

a. EFW Initiation;

b. EFW Vector Valve Control;

c. Main Steam Line Isolation; and

d. Main Feedwater (MFW) Isolation.

EFW is initiated to restore a source of cooling water to the secondary system when conditions indicate that the normal source of feedwater is insufficient to continue heat removal. The two indications used for this are the loss of both MFW pumps and a low level in the steam generator (SG). Also, EFW is initiated when action is being taken to isolate the MFW from the SG during conditions of uncontrolled depressurizations. This is done by initiating EFW when steam pressure reaches the low SG pressure setpoint for isolation of main steam and MFW, and EFW vector valve control. Finally, EFW is initiated when the primary system experiences a total loss of forced circulation. This initiation, on the loss of all reactor coolant pumps (RCPs),

(continued)

INSERT B3.3-91A

address containment overpressurization concerns by isolating main feedwater (MFW) to both steam generators during an MSLB and to mitigate core overcooling concerns.

Steam generator header pressure is used as input signals to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The turbine-driven emergency feedwater (TDEFW) pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. These functions are credited for mitigating an MSLB. The function of closing the main and startup feedwater block valves is not credited in the MSLB analysis for mitigation of containment overpressurization during a MSLB. However, the MSLB detection and MFW isolation circuitry performs this function.

There are three pressure transmitters per steam generator with each feeding a steam pressure signal to a signal isolator (when used) and bistable. These bistables are calibrated to provide an ON/OFF signal at the desired setpoint for actuation of the feedwater isolation circuitry. A pressure transmitter and its associated signal isolator(s) and bistable(s) constitute a MSLB detection analog channel.

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the digital channels will not result in an in advertent feedwater isolation. If either digital channel is actuated, an MFW isolation will occur. Energizing the master relay results in closure of contacts in various control circuits for systems and components used for the MSLB containment overpressurization protection. Therefore, when the master relay is energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. The two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, time delay, and test/manual actuation pushbutton are considered a feedwater isolation digital channel.

The feedwater isolation digital channels are enabled and disabled administratively rather than automatically. Appropriate operating procedures contain provisions to enable/disable the digital channels.

(4) <Entire Page>

BASES

BACKGROUND (continued)

ensures the EFW is available to raise SG levels to promote natural circulation cooling. Additionally, this ensures that EFW is available under the worst-case, small break loss of coolant accident (LOCA) conditions when secondary system cooling with high SG water levels is necessary.

The EFIC System also isolates main steam and MFW to an SG that has lost pressure control. With the loss of pressure control, the heat sink temperature control is lost and the heat removal rate cannot be controlled. The main steam and MFW are isolated to an SG when the steam pressure reaches a low setpoint, a condition which is beyond the normal operating point of the secondary system.

The EFIC System also performs an EFW control function to avoid delivering EFW to a depressurized SG when the other SG remains pressurized. This continues the function of isolating functional components from an SG whose pressure cannot be controlled. This function precludes the delivery of fluid to a depressurized SG, thereby avoiding an uncontrolled cooling condition as long as the other SG remains pressurized. When both of the SGs are depressurized, the EFIC logic provides EFW flow to both SGs until a significant pressure difference between the two SGs is developed, thereby ensuring that core cooling is maintained.

Trip Setpoints and Allowable Values

The trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± [rack calibration + comparator setting accuracy]).

The trip setpoints used in the bistables are based on the analytical limits stated in FSAR, Section [14.1] (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. The Allowable Values specified in Table 3.3.11-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits to allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environmental errors for those ~~EFIC channels~~ that must function in harsh

(continued)

4

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

environments as defined by 10 CFR 50.49 (Ref. 2). A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in "[Unit Specific Setpoint Methodology]" (Ref. 3). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actuation trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the consequences of Design Basis Accidents (DBAs) are acceptable, providing the unit is operated from within the LCOs at the onset of the DBA, and that the equipment functions as designed.

Each channel can be tested on line to verify that the setpoint accuracy is within the specified allowance requirements of Figure [ ], FSAR, Chapter [7] (Ref. 4). Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. The SRs for the channels are specified in the SRs Section.

The Allowable Values listed in Table 3.3.11-1 are based on the "[Unit Specific Setpoint Methodology]" (Ref. 3), which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Figure [ ], FSAR, Chapter [7] (Ref. 4), illustrates EFIC EFW Initiation logic operation.

Each EFIC train actuates on a one-out-of-two taken twice combination of trip signals from the instrumentation channels. Each EFIC channel can issue an initiate command, but an EFIC actuation will take place only if at least two

(continued)

4 <Entire Page>

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

channels issue initiate commands. The one-out-of-two taken twice logic combinations are transposed between trains so that failure of two channels prevents actuation of, at most, one train.

More detailed descriptions of the EFIC instrumentation are provided next.

1. EFW Initiation

Figure [ ], FSAR, Chapter [7] (Ref. 4), illustrates one channel of the EFIC EFW Initiation channel. The individual instrumentation channels that serve EFIC EFW Initiation Function are discussed next.

a. Loss of MFW Pumps (Control Oil Pressure)

Loss of both MFW Pumps is one of the four parameters within the EFIC System that automatically initiates EFW. Loss of MFW Pumps is detected by MFW Pump turbine control oil pressure. The MFW Pump status instrumentation is a part of the nuclear instrument (NI) and Reactor Protection System (RPS). Each RPS channel receives MFW Pump status information from pressure switches (four per pump). If both switches in a single channel trip, the associated RPS channel trips. Each RPS channel provides both MFW Pumps tripped signal to the associated EFIC channel. The trip Function is bypassed when THERMAL POWER ≤ 20% RTP and the RPS is in shutdown bypass. The bypass is automatically removed when THERMAL POWER is greater than 20% RTP.

Loss of both MFW Pumps was chosen as an EFW automatic initiating parameter because it is a direct and immediate indicator of loss of MFW.

b. SG Level - Low

Four EFIC dedicated low range level transmitters per SG Level - Low are used to generate the signals used for detection for low level

(continued)

④ <Entire Page>

BASES

BACKGROUND

b. SG Level - Low (continued)

conditions for EFW actuation. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control SG level at the low level setpoint [30 inches] when one or more RCPs are operational.

The lower and upper taps for the low range level transmitters are located at 6 inches and 277 inches, respectively, above the upper face of the SG's lower tube sheet. The calibrated range is 0-150 inches.

SG Level - Low was chosen as an EFW automatic initiating parameter because it indicates that the primary feedwater source is insufficient to meet the heat removal requirements and, therefore, additional cooling water is necessary to ensure core decay heat removal.

c. SG Pressure - Low

Four transmitters per SG provide the EFIC System with channels A through D of SG Pressure - Low. These are the same transmitters used by the MFW and Main Steam Line Isolation Functions. When the SG pressure drops below the bistable setpoint of 600 psig on a given channel, an EFW Initiation signal is sent to the automatic actuation logic. The low pressure Function may be manually bypassed when both SGs are less than 750 psig. If either SG input channel exceeds 750 psig, the EFIC channel bypass is automatically removed. The low pressure operational bypass allows for normal cooldown without EFIC actuation.

SG Pressure - Low is a primary indication and actuation signal for steam line breaks (SLBs) or feedwater line breaks (FWLBs). For small breaks, which do not depressurize the SG or take a long time to depressurize, automatic actuation is not required. The operator has time to diagnose the problem and take the appropriate actions.

(continued)

④ <Entire Page>

BASES

BACKGROUND (continued)

d. RCP Status

A loss of power to all four RCPs is an indication of a pending loss of forced flow in the Reactor Coolant System. These sensing signals are input into the four channels of EFIC.

When at least two channels issue initiate commands based on loss of all RCPs, the EFIC System will automatically actuate EFW and switch the level control setpoint to approximately 50% in the SG. This higher setpoint provides a thermal center in the SG at a higher elevation than that of the reactor to ensure natural circulation of the reactor coolant.

To allow heatup and cooldown operations without actuation, a bypass permissive of 10% RTP is used. The 10% bypass permissive was chosen because it was an available, qualified Class 1E signal at the time the EFIC System was designed. When the first RCP is started, the "loss of four RCPs" initiation signal may be manually reset. If the bypass is not manually reset, it will be automatically reset when the unit reaches 10% power. During cooldown, the bypass may be inserted at any time the power has been reduced below 10%. However, for most operating conditions, it is recommended that this trip function remain active until after the Decay Heat Removal System has been initiated and the system is ready for the last RCP to be tripped. This trip function must be bypassed prior to stopping the last RCP.

2. EFW Vector Valve Control

Figure [ ], FSAR, Chapter [7] (Ref. 4), illustrates one channel of the EFIC EFW Vector Valve Control logic. The function of the EFW vector logic is to determine whether EFW should not be fed to one or the other SG. This is to preclude the continued addition of EFW to a depressurized SG and, thus, to minimize the overcooling effects of a steam leak.

(continued)

4

BASES

BACKGROUND 2. EFW Vector Valve Control (continued)

Each set of vector logic receives SG pressure information from bistables located in the input logic of the same EFIC channel. The pressure information received is:

a. SG A pressure less than 600 psig;

b. SG B pressure less than 600 psig;

c. SG A pressure 125 psid greater than SG B pressure; and

d. SG B pressure 125 psid greater than SG A pressure.

Each vector logic also receives a vector/control enable signal from both EFIC channel A and channel B when EFW is initiated. [Each logic also receives an SG high level signal. High level in an SG prevents opening the associated vector valves and enables closing the valves without either EFIC train vector valve enable.]

The vector logic develops signals to open or to close SG A and B EFW valves.

The vector logic outputs are in a neutral state until enabled by the control/vector enable from the channel A or B trip logics. When enabled, the vector logic can issue open or close commands to the EFW control valves and EFW isolation valves per the selected channel assignments.

Each vector logic may isolate EFW to one SG or the other, never both.

(continued)

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BASES

BACKGROUND

2. EFW Vector Valve Control (continued)

The valve open or close commands are determined by the relative values of SG pressures as follows:

	SG VALVES	
PRESSURE STATUS	"A"	"B"
SG A and SG B > 600 psig	Open	Open
| SG A - SG B | < 125 psid	Open	Open
SG A or SG B ≤ 600 psig and SG A - SG B ≥ 125 psid	Open	Close
SG A or SG B ≤ 600 psig and SG B - SG A ≥ 125 psid	Close	Open

Bypass

One of the four initiation channels can be put into "maintenance bypass." Bypassing one initiation channel isolates that channel's signal to the functions fed from initiation channel but does not bypass the trip logic within the actuation channel. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC System receives signals from NI and RPS, the maintenance bypass from the NI and RPS is interlocked with the EFIC System. If one channel of the NIS and RPS is in maintenance bypass, only the corresponding channel of the EFIC may be bypassed (e.g., channel A, NI or RPS, and channel A, EFIC). This ensures that only the corresponding channels of the EFIC and NI and RPS are placed in maintenance bypass at the same time.

EFIC channel maintenance bypass does not bypass EFW Initiation from Engineered Safety Feature Actuation System (ESFAS) high pressure injection (HPI). The EFIC HPI Actuation Function is, however, bypassed when ESFAS is bypassed.

The operational bypass provisions were discussed as part of the individual Functions described earlier.

(continued)

4

BASES

BACKGROUND

Bypass (continued)

Operational bypass of the OTSG Level - High input to the vector valve logic is possible after EFIC initiation. [For this unit, bypassing the overfill function is for the following reasons:]

3, 4. Main Steam Line and MFW Isolation

Figure [ ], FSAR, Chapter [7] (Ref. 4) illustrates one channel of the EFIC Main Steam Line and MFW Isolation logic. Four pressure transmitters per SG provide EFIC with channels A through D logic of SG pressure. The channels are as described for EFW Initiation mentioned earlier.

Once isolated, manual action is required to defeat the isolation command if desired. The EFIC System is designed to perform its intended function with one channel in maintenance bypass (in effect, inoperable) with a single failure in one of the remaining channels. This is in compliance with IEEE-279-1971 (Ref. 5) due to the redundancy and independence in the EFIC design.

APPLICABLE SAFETY ANALYSES

INSERT B 3.3-99 A

4

1. EFW Initiation

Although loss of both MFW pumps is a direct and immediate indicator of loss of MFW, other scenarios such as valve closures could potentially cause loss of feedwater. The loss of MFW analysis, therefore, conservatively assures the actuation of EFW on low SG level. If the loss of feedwater is due to loss of MFW pumps, EFW will be actuated much earlier than assumed in the analysis, which will increase the SG heat transfer capability and will lessen the severity of the transient.

The DBA which forms the basis for initiation of the EFW systems is a loss of MFW transient. In the analysis of this transient, SG Level - Low is the parameter assumed to automatically initiate EFW. This assumption yields the least SG inventory available for heat removal and is, therefore, conservative for

(continued)

INSERT B 3.3-99A

Based on the containment pressure response reanalysis, the containment design pressure would be exceeded for a MSLB inside containment without operator action to isolate main feedwater and installed equipment necessary to automatically isolate main feedwater to both steam generators during a MSLB.

Steam generator header pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs are closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for decay heat removal. All of these functions are credited for mitigating a MSLB inside containment.

~~BASES~~

~~APPLICABLE SAFETY ANALYSES~~

1. EFW Initiation (continued)

evaluation of this DBA. SG Level - Low would be an indicator of all accidents involving a loss of primary to secondary heat removal.

SG Pressure - Low is a primary indication and provides the actuation signal for SLBs or FWLBs. For small breaks, which do not depressurize the SG or take a long time to depressurize, automatic actuation is not required. The operator has sufficient time to diagnose the problem and take the appropriate actions.

Loss of four RCPs is a primary indicator of the need for auxiliary feedwater (AFW) in the safety analyses for loss of electric power and loss of coolant flow. It also serves as a backup indicator for SLBs and small break LOCAs.

4

2. EFW Vector Valve Control

Most of the FSAR SLB analyses were performed prior to the development of the safety grade EFIC System. Therefore, the EFIC vector valve control was not credited in the original licensing basis for a main SLB analysis. Instead, operator action was credited with isolating AFW to the affected SG within the first 60 seconds. However, isolating the affected SG is a function automatically performed by the EFIC System. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve control.

3, 4. Main Steam Line and MFW Isolation

The FSAR analysis assumed integrated control system action for MFW and Main Steam Line Isolation. The analysis took credit for turbine stop valve closure and feedwater valve isolation on reactor trip and considered the isolation functions occurring on SG pressure < 600 psig as backup. These isolation functions are currently provided by the safety grade EFIC System. Use of the EFIC System in the original safety analysis would have been consistent with the licensing position allowing mitigative functions to be

(continued)

5 <Except as marked>

BASES

APPLICABLE SAFETY ANALYSES

~~3. 4. Main Steam Line and MFW Isolation (continued)~~

~~performed by safety grade systems in accident analysis. For these reasons, the SLB accident analysis remains conservative with the assumed integrated control system actions.~~

A

The ~~EFIC System~~ Instrumentation satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3).

4

LCO

This LCO requires that instrumentation necessary to initiate a MFW isolation

~~All instrumentation performing an EFIC System Function in Table B 3.3.11-1 shall be OPERABLE.~~ Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the ~~affected~~ Functions.

~~Four~~ Three channels per SG are required to be OPERABLE ~~for all EFIC instrumentation channels~~ to ensure that no single failure prevents MFW isolation ~~actuation of a train~~. Each ~~EFIC~~ instrumentation channel ~~is considered to~~ includes the sensors and measurement channels ~~for each Function, the operational bypass switches, and permissives. Failures that disable the capability to place a channel in operational bypass, but which do not disable the trip Function, do not render the protection channel inoperable.~~

A

~~Only the Allowable Values are specified for each EFIC initiation and bypass removal function in the LCO. In Table 3.3.11-1, Allowable Values for the bypass removal functions are specified in terms of applicability limits on the associated trip Function. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "[Unit Specific Setpoint Methodology]" (Ref. 3).~~

(continued)

BASES

5

LCO (continued)

The Bases for the LCO requirements of each specific EFIC Function are discussed next.

Loss of MFW Pumps

Four EFIC channels shall be OPERABLE with MFW pump turbines A and B control oil low pressure actuation setpoints of > [55] psig. The 55 psig setpoint is about half of the normal operating control oil pressure. The 55 psig setpoint Allowable Value was arbitrarily chosen as a good indication of Loss of MFW Pumps. Analysis only assumes Loss of MFW Pumps and a specific value of MFW pump control oil pressure is not used in the analysis. The Loss of MFW Pumps Function includes a bypass enable and removal function from the NI/RPS. The bypass removal function is based on maintaining consistency with RPS LCO and design of system.

SG Level - Low

Four EFIC dedicated low range level transmitters per SG shall be OPERABLE with SG Level - Low actuation setpoints of ≥ [9] inches, to generate the signals used for detection for low level conditions for EFW Initiation. There is one transmitter for each of the four channels A, B, C, and D. The signals are also used after EFW is actuated to control at the low level setpoint of 30 inches when one or more RCPs are in operation. In the determination of the low level setpoint, it is desired to place the setpoint as low as possible, considering instrument errors, to give the maximum operability margin between the integrated control system low load control setpoint and the EFW Initiation setpoint. This will minimize spurious or unwanted initiation of EFW. Credit is only taken for low level actuation for those transients which do not involve a degraded environment. Therefore, normal environment errors only are used for determining the SG Level - Low level setpoint.

SG Pressure - Low

Four EFIC channels per SG shall be OPERABLE with SG low pressure actuation setpoints of ≥ [600] psig. The setpoint is chosen to avoid actuation under transient conditions not requiring secondary system isolation, preferring to maintain

(continued)

5

BASES

LCO

SG Pressure - Low (continued)

a steaming path to the condenser, if possible. Small break LOCA analyses have indicated minimum secondary system pressures of approximately 700 psig. The SG Pressure - Low Function includes a bypass enable and removal function. The bypass removal Allowable Value is chosen to allow sufficient operating margin for the operator to bypass when cooling down.

SG Differential Pressure - High

Four EFIC channels for SG differential pressure shall be OPERABLE with setpoints of ≤ [125] psid. The setpoint ensures that automatic EFW isolation to a depressurized SG occurs for the range of sizes of SLBs that require rapid actuation early in the event. The setpoint has also been chosen to avoid spurious isolation of EFW during conditions due to relatively small deviations in SG pressures that can be caused by primary system conditions. The SG Differential Pressure - High Function includes a bypass enable and removal function. The bypass removal Allowable Value is chosen to allow sufficient operating margin for the operator to bypass when cooling down.

RCP Status

Four EFIC channels for RCP status shall be OPERABLE. This ensures that upon the loss of four RCPs, EFW will be automatically initiated with the EFW control level automatically raised to approximately 50%, providing a higher SG level for establishing and maintaining natural circulation conditions when the forced reactor coolant flow is lost. No setpoint is specified since the status indication as used by EFIC is binary in nature. The RCP Status Function includes a bypass enable and removal function from the RPS. The Allowable Value for the bypass removal is set high enough to avoid spurious actuations during low power operation.

(continued)

A -2

4 <except as marked>

BASES

LCO (continued)

SG Level - High

For this unit, the basis for SG Level - High signal is as follows:

5

APPLICABILITY

The EFIC System instrumentation Functions shall be OPERABLE in accordance with Table 3.3.11-1. Each Function has its own requirements that are based on the specific accidents and conditions that it is designed to protect against.

The initiation of EFW on the Loss of MFW Pumps shall only be required in MODE 1 and in MODES 2 and 3 when not in shutdown bypass, when core power production and heat removal requirements are the greatest. Below these unit conditions, the EFW Initiation on low SG level is rapid enough to avoid unnecessary primary system overheating.

EFW Initiation on low SG level shall be OPERABLE at all times the SG is required for heat removal. These conditions include MODES 1, 2, and 3. To avoid automatic actuation of the EFW pumps during normal heatup and cooldown transients, the low SG pressure Function can be bypassed at or below a secondary pressure of [750] psig. This secondary pressure can normally only be reached during MODE 3 operation.

The EFW System initiation on loss of all RCPs Function shall be operable at ≥ 10% RTP. It is possible to bypass the Function below 10% RTP; however, for most cases, the Function is kept in service until the unit is placed on the Decay Heat Removal System. To prevent inadvertent actuation of the EFW pumps, it must be bypassed prior to stopping the last RCP.

5

MSLB Detection and

The MFW, ~~Main Steam Line~~ Isolation ~~and EFW Vector Valve Control~~ Function~~s~~ shall be OPERABLE in MODES 1, and 2, and 3 MODE with main steam header ~~SG~~ pressure ≥ 700 ~~750~~ psig because the SG inventory can be at a high energy level and contribute significantly to the peak pressure with a secondary side break. ~~Both the normal~~ main feedwater ~~and the EFW~~ must be able to be isolated on each SG to limit ~~overcooling of the primary and~~ mass and energy releases to the reactor building. Once the SG pressures have decreased below 700 ~~750~~ psig, the ~~Main Steam Line and~~ MFW Isolation Function~~s~~ can be bypassed to avoid actuation during normal unit cooldowns. ~~The EFW Vector Valve Control~~ 5

INSERT B 3.3-104A

(continued)

INSERT B 3.3-104A

Also during MODE 3 the MFW isolation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since the function of the instrumentation is already fulfilled.

5 <Entire Page>

BASES

APPLICABILITY (continued)

~~logic will not perform any function when both SG pressures are low; thus, the logic can also be bypassed at the same point.~~ In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent. In MODES 4, 5, and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy and ~~EFIC~~ MSLB Detection and MFW Isolation instrumentation is not required to be OPERABLE.

ACTIONS

If a channel's trip setpoint is found nonconservative with respect to the Allowable Value, or any of the transmitter or signal processing electronics ~~or EFIC channel cabinet modules~~ are found inoperable, then ~~all affected~~ the Functions provided by that channel must be declared inoperable and the unit must enter the appropriate Conditions ~~for the particular protection Function affected~~.

A Note has been added to the ACTIONS indicating that a separate Condition entry is allowed for each ~~Function~~ SG header (MFW Isolation Function).

A.1 ~~and A.2~~

Condition A applies to failures of instrumentation channels associated with a single ~~EFW Initiation, Main Steam Line Isolation, or~~ MSLB Detection and MFW Isolation instrumentation channel in one or more MFW Isolation Functions. ~~This includes failure of a common instrumentation channel in any combination of the Functions.~~

With one channel inoperable in one or more ~~EFW Initiation, Main Steam Line Isolation, or~~ MFW Isolation Function~~s listed in Table 3.3.11-1~~, the channel(s) must be placed in ~~bypass or~~ trip within ~~1~~ 4 hours. ~~This Condition applies to failures that occur in a single channel, e.g., channel A, which when bypassed will remove initiate Functions within the channel from service. Since the RPS and EFIC channels are interlocked, only the corresponding channel in each system may be bypassed at any time. This feature is ensured by an electrical interlock. If testing of another channel in either the EFIC or RPS is required, the EFIC channel must be placed in trip to allow the other channel to be bypassed. With the channel in trip, the resultant logic is one-out-of-two.~~ Tripping the affected channel places the Function in a one-out-of-two configuration. Operation in this configuration may continue indefinitely since the MSLB Detection and MFW Isolation Function is capable of performing its isolation function in the presence of any single random failure. The Completion Time of ~~1~~ 4 hour~~s~~ is adequate to perform Required Action A.1.

(continued)

BASES

ACTIONS

5

A.1 and A.2 (continued)

Required Action A.2 provides for placing the channel(s) in trip if the channel(s) is/are not restored to OPERABLE status within 72 hours.

A single inoperable EFIC instrumentation channel affects at most one train of EFW, Main Steam Line Isolation, and MFW Isolation. Therefore, the 72 hour Completion Time was selected to be consistent with the allowed out of service time for the EFW, Main Steam Line Isolation, and MFW Isolation Functions.

B.1, B.2, and B.3

Condition B applies to a situation where two instrumentation channels for multiple protection functions of EFW Initiation, Main Steam Line Isolation, or MFW Isolation instrumentation are inoperable. For example, Condition B applies if channel A and B of the EFW Initiation Function are inoperable.

Condition B does not apply if one channel of different Functions is inoperable in the same protection channel. That condition is addressed by Condition A.

With two EFW Initiation, Main Steam Line Isolation, or MFW Isolation protection channels inoperable, one channel must be placed in bypass (Required Action B.1). Bypassing one of the remaining OPERABLE channels is not possible due to system interlocks. Therefore, the second channel must be tripped (Required Action B.2) to prevent a single failure from causing loss of the EFIC Function. The Completion Times of 1 hour are adequate to perform the Required Actions.

One of the channels must be returned to OPERABLE status (Required Action B.3) to minimize the time the system is permitted to operate in a configuration that is not capable of withstanding a single failure and still initiate EFW, Main Steam Line Isolation, and MFW Isolation. Restoring one channel changes system status to that of Condition A. A single inoperable EFIC channel affects at most one train of EFW, Main Steam Line Isolation, and MFW Isolation. Therefore the 72 hour Completion Time was selected to be

(continued)

5 <Entire Page>

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

consis~~tent~~ with the all~~owed out~~ of service time for the ~~EFW,~~ Main Steam Line Isolation, and MFW Isolation ~~Functions~~.

C.1

The function of the EFW Vector Valve Control is to meet the single-failure criterion while being able to provide EFW on demand and isolate an SG when required. These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system. Refer to LCO 3.3.14, "Emergency Feedwater Initiation and Control (EFIC) Emergency Feedwater (EFW) – Vector Valve Logic."

With one EFW Vector Valve Control channel inoperable, the system cannot meet the single-failure criterion and still meet the dual functional criteria described earlier. This condition is analogous to having one EFW train inoperable. Therefore, when one vector valve control channel is inoperable, the channel must be restored to OPERABLE status (Required Action C.1) within 72 hours, which is consistent with the Completion Time associated with the loss of one train of EFW.

with two channels in one or more MSLB Detection and MFW Isolation Function inoperable or

B.1, B.2.1, and B.2.2
~~D.1, D.2.1, D.2.2, E.1, and F.1~~

and associated Completion Time of Condition A

MODE 3 with 12 hours and

~~If~~ the Required Actions ~~cannot be~~ met ~~within the required Completion Time~~, the unit must be placed in ~~a MODE or condition in which the requirement does not apply. This is done by placing the unit in a nonapplicable MODE for the particular Function. The nonapplicable MODE is to open the CRD trip breakers for Function 1.a, MODE 4 for Function 1.b, less than 10% RTP for Function 1.d, and~~ SG pressure ~~less than~~ ~~750~~ 700 psig ~~for all other Functions~~. The allowed Completion Time~~s~~ ~~are~~ is reasonable, based on ~~operating experience~~, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

must be reduced to

main steam header

or all MFCVs and SFCVs must be closed within 18 hours.

(continued)

BASES (continued)

5

SURVEILLANCE REQUIREMENTS

~~A Note indicates that the SRs for each EFIC instrumentation Function are identified in the SRs column of Table 3.3.11-1. All Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION. The SG - Low Level Function is the only Function that was modeled in transient analysis, and thus is the only EFW Initiation Function subjected to response time testing. Response time testing is also required for Main Steam Line and MFW Isolation. Individual EFIC subgroup relays must also be tested, one at a time, to verify the individual EFIC components will actuate when required. Some components cannot be tested at power since their actuation might lead to unit trip or equipment damage. These are specifically identified and must be tested when shut down. The various SRs account for individual functional differences and for test frequencies applicable specifically to the Functions listed in Table 3.3.11-1. The operational bypasses associated with each EFIC instrumentation channel are also subject to these SRs to ensure OPERABILITY of the EFIC instrumentation channel.~~

SR 3.3.11.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION.

2

Agreement criteria are ~~determined by the unit staff~~ based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the

(continued)

4 <Except as marked>

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.1 (continued)

criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified, where practical, ~~to be~~ reading at the bottom of the range and not failed downscale. 2

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel operability during normal operational use of the displays ~~associated~~ with the LCO required channels. 2

~~SR 3.3.11.2~~

~~A CHANNEL FUNCTIONAL TEST verifies the function of the required trip, interlock, and alarm functions of the channel. Setpoints for both trip and bypass removal functions must be found within the Allowable Value specified in the LCO. (Note that the Allowable Values for the bypass removal functions are specified in the Applicable MODES or Other Specified Condition column of Table 3.3.11-1 as limits on applicability for the trip Functions.) Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.~~ 5

~~The Frequency of 31 days is based on unit operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.~~

SR 3.3.11.~~3~~2 5

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.11.~~3~~ 2 5 (continued)

and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the ~~unit specific~~ setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the ~~unit specific~~ setpoint analysis. 2

The Frequency is based on the assumption of an [18] month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. 1

SR 3.3.11.4 5

~~This SR verifies individual channel actuation response times are less than or equal to the maximum value assumed in the accident analysis.~~

~~Response time testing acceptance criteria are included in "Unit Specific Response Time Acceptance Criteria" (Ref. 6).~~

~~Individual component response times are not modeled in the analysis. The analysis models the overall or total elapsed time, from the point at which the parameter exceeds the actuation setpoint value at the sensor, to the point at which the end device is actuated.~~

~~EFIC RESPONSE TIME tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the EFIC RESPONSE TIME, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month test Frequency is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences. EFIC RESPONSE TIMES cannot be determined at power since equipment operation is required.~~

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [14.1]. -4
2. 10 CFR 50.49.
3. [Unit Name], "[Unit Specific Setpoint Methodology]."
4. FSAR, Chapter [7].
5. IEEE-279-1971, April 1972.
6. "Unit Specific Response Time Acceptance Criteria."

-4

1. 10 CFR 50.36. -4

2 - A <is used for inserting this text on subsequent pages> MSLB Detection and MFW Isolation - 2

4 <Except as marked>

Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation - 2

B 3.3 INSTRUMENTATION

B 3.3.12 ~~Emergency Feedwater Initiation and Control (EFIC)~~ Manual Initiation

BASES

A

BACKGROUND

The ~~EFIC~~ manual initiation ~~capability~~ the isolation provides the operator with the capability to actuate ~~EFIC~~ Function~~s~~ from the control room. ~~in the absence of any other initiation condition.~~ ~~Manually actuated Functions include main feedwater (MFW) Isolation for once through steam generator (SG) A, MFW Isolation for SG B, Main Steam Line Isolation for SG A, Main Steam Line Isolation for SG B, and Emergency Feedwater (EFW) Actuation.~~ Th~~ese~~is Function~~s~~ ~~are~~ is provided in the event the operator determines that ~~an EFIC~~ the Function is needed and does not automatically actuate. Th~~ese~~ is ~~are~~ backup Function~~s~~ to ~~those performed~~ the automatic~~ally by EFIC~~. is a MFW isolation

A The ~~EFIC~~ manual initiation circuitry satisfies the manual initiation and single-failure criterion requirements of IEEE-279-1971 (Ref. 1).

APPLICABLE SAFETY ANALYSES

~~EFIC Functions~~ The MFW Isolation Function credited in the safety analysis ~~are~~ is automatic. However, the manual initiation Function~~s~~ ~~are~~ is required by design as backup~~s~~ to the automatic ~~trip~~ the Function~~s~~ and allows operators to actuate ~~EFW, Main Steam Line Isolation, or~~ MFW Isolation whenever th~~ese~~is Function~~s~~ ~~are~~ is needed. Furthermore, the manual initiation of ~~EFW Actuation, Main Steam Line Isolation, and~~ MFW Isolation may be specified in unit operating procedures.

A The ~~EFIC~~ manual initiation function~~s~~ satisf~~y~~ Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2).

LCO

~~All instrumentation performing an EFIC manual initiation Function shall be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions.~~

~~Two~~ One manual initiation switch~~es~~ per actuation channel (A and B) ~~of each Function (A and B MFW Isolation, A and B Main Steam Line Isolation, and EFW Actuation) are~~ is required to be OPERABLE. ~~whenever the SGs are being relied on to remove~~

5

(continued)

5 < Except as marked >

A - 2

BASES

function

The

LCO (continued)

~~heat. Each Function~~ (MFW Isolation, ~~Main Steam Line Isolation, and EFW Initiation)~~ has two actuation or "trip" channels, channels A and B. Within each channel ~~the~~ actuation logic there ~~are two~~ manual trip switch~~es~~. When ~~one~~ The manual switch is depressed, ~~a half trip occurs. When both~~ manual ~~switches are depressed,~~ a full trip of channel A actuation or B occurs. ~~for that particular Function.~~ ~~Similarly, channel B actuation logic for each Function has two manual trip switches. Both switches per actuation channel must be OPERABLE and must be depressed to get a full manual trip of that channel. The use of two manual trip switches for each channel of actuation logic allows for testing without actuating the end devices and also reduces the possibility of accidental manual actuation.~~

is one

and

with main steam header pressure ≥ 700 psig

MODE

APPLICABILITY

The MFW ~~and Main Steam Line~~ Isolation manual initiation Function~~s~~ shall be OPERABLE in MODES 1, 2, and 3 because SG inventory can be at a sufficiently high energy level to contribute significantly to the peak containment pressure during a secondary side break. In MODES 4, 5, and 6, the SG energy level is low and secondary side feedwater flow rate is low or nonexistent.

INSERT B 3.3-113A

~~The EFW manual initiation Function shall be OPERABLE in MODES 1, 2, and 3 because the SGs are relied on for Reactor Coolant System heat removal. In MODES 4, 5, and 6, heat removal requirements are reduced and can be provided by the Decay Heat Removal System.~~

ACTIONS

~~A Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each EFIC manual initiation Function.~~

~~A.1~~

~~With one or both manual initiation switches of one or more EFIC Function(s) inoperable in one channel, the channel for the associated EFIC Function(s) must be placed in the tripped condition within 72 hours. With the channel in the tripped condition, the single-failure criterion is met and the operator can still initiate one actuation channel given~~

(continued)

INSERT B 3.3-113A

During MODE 3, the MFW Isolation manual initiation Function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled.

(5) <Entire Page>

BASES

ACTIONS

A.1 (continued)

a single failure in the other channel. Failure to perform Required Action A.1 could allow a single failure of another switch to prevent manual actuation of at least one of two trip channels. The Completion Time allotted to trip the channel allows the operator to take all the appropriate actions for the failed channel and still ensure that the risk involved in operating with the failed channel is acceptable.

A.~~B~~.1

INSERT B3.3-114A

With one or both manual initiation switches ~~of one or more EFIC Function(s) inoperable in both actuation channels~~, one actuation channel for each Function must be restored to OPERABLE status within 1 hour. With the channel restored, the second channel must be placed in the tripped condition within 72 hours (Required Action A.1). With the channel in the tripped condition, the single-failure criterion is met and the operator can still initiate one actuation channel given a single failure in the other channel. The Completion Time allotted to restore the channel allows the operator to take all the appropriate actions for the failed channel and still ensures that the risk involved in operating with the failed channel is acceptable.

B.1
~~C.1 and C.2~~

With both manual initiation switches inoperable or the

and associated Completion Time of Condition A not met,

~~If~~ Required Action A.1 or Required Action B.1 ~~cannot be met within the required Completion Time~~, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

INSERT B3.3-114B

(continued)

INSERT B 3.3-114A

inoperable, the manual initiation switch must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and MFW Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and MFW Isolation Function.

INSERT B 3.3-114B

placed in MODE 3 within 12 hours and the main steam header pressure reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.12.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. ~~However, for MFW and Main Steam Line Isolation, the test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated with MFW and Main Steam Line Isolation or actuating EFW during testing at power. The Frequency of 31 days is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.~~ 4

INSERT B33-115A

REFERENCES

1. IEEE-279-1971, April 1972.
2. 10 CFR 50.36. 2

INSERT B 3.3-115A

The Frequency of 18 months is based on engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during unit operation is avoided.

Main Steam Line Break (MSLB) Detection and Main Feedwater (MFW) Isolation 2

B 3.3 INSTRUMENTATION

B 3.3.13 ~~Emergency Feedwater Initiation and Control (EFIC)~~ Logic Channels

BASES

BACKGROUND

4 INSERT B3.3-116A

~~Main Steam Line and Main Feedwater (MFW) Isolation~~

~~The four emergency feedwater initiation and control (EFIC) channels sensing a steam generator (SG) low outlet pressure condition input their initiate commands to the trip logic modules. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates the Main Steam Line and MFW Isolation Logics. The trip logic modules are physically located in the "A" and "B" EFIC channel cabinets. Channel "A" actuation logic initiates when instrumentation channel "A" or "B" initiates and channel "C" or "D" initiates, which in simplified logic is:~~

~~"A" actuation = (A and C) or (A and D) or (B and C) or (B and D)~~

~~Channel "B" actuation logic initiates when instrumentation channel "A" or "C" initiates and channel "B" or "D" initiates, which in simplified logic is:~~

~~"B" actuation = (A and B) or (A and D) or (C and B) or (C and D)~~

~~Each of the four Functions (SG A Main Feedwater Isolation, SG B Main Feedwater Isolation, SG A Main Steam Line Isolation, and SG B Main Steam Line Isolation) has a channel "A" and a channel "B" of automatic actuation logic.~~

~~Both channels "A" and "B" of the SG A Main Feedwater Isolation automatic actuation logic send closure signals to the SG A main feedwater pump suction valve, the three SG A block valves, and the MFW pump discharge cross connect valve. In addition, the instrumentation trips MFW pump "A."~~

~~Both channels "A" and "B" of the SG A Main Steam Line Isolation automatic actuation logic send closure signals to both of the SG A Main Steam Isolation valves.~~

~~SG B MFW and Main Steam Line Isolation automatic actuation logics respond similarly for the SG B valves and MFW pump "B."~~

(continued)

INSERT B 3.3-116A

The six MSLB detection analog channels feed two redundant feedwater isolation digital channels consisting of two single failure proof two-out-of-three logic circuits. If the logic is satisfied, a master relay coil is energized. The use of an energized master relay ensures that a loss of power to the digital channels will not result in an inadvertent feedwater isolation. If either digital channel is actuated, an MFW isolation will occur. Energizing the master relay results in closure of contacts in various control circuits for systems and components used for the MSLB containment overpressurization protection. Therefore, when the master relay is energized, the systems and components perform their isolation functions. Other features of the digital channels include a test/manual actuation pushbutton, a circuit seal-in after the master relay is energized, a 2 second time delay to prevent spurious actuation, and an "enable" or "arming" switch. Each of the two two-out-of-three logic circuits, along with their associated enable switch, master relay, seal-in, and time delay is considered a feedwater isolation digital channel.

4

BACKGROUND (continued)

Emergency Feedwater (EFW) Actuation

The four EFIC instrumentation channels for each of the parameters being sensed input their initiate commands to the trip logic modules. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates the EFW initiation logic. These trip logic modules are physically located in the "A" and "B" EFIC channel cabinets.

EFW Actuation functions are the same logic combinations as MFW and Main Steam Line Isolation. EFW initiation also occurs on high pressure injection (HPI) initiation. Both trains of HPI initiation are input into each EFW initiate logic channel.

EFIC automatically initiates the EFW System when any of the following conditions exist:

a. All four reactor coolant pumps are tripped;

b. Both MFW pumps are tripped and reactor power is > 20% RTP with the nuclear instrumentation Reactor Protection System not in shutdown bypass;

c. Low level in either once through SG;

d. Low pressure in either SG; or

e. HPI Actuation on both A and B Engineered Safety Feature Actuation System channels.

Vector Valve Enable Logic

The EFW module logic is responsible for sending open or close signals to the EFW control and isolation valves. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates the vector valve logic. The vector module logic outputs are in a neutral state (neither commanding open nor close) until a signal is received from the Vector Valve Enable Logic. The Vector Valve Enable Logic monitors the channel A and B EFW Actuation logics. When an EFW Actuation occurs, the vector enable logic enables the vector logic to generate open or close signals to the EFW valves depending on the relative values of SG pressures.

(continued)

5 <Except as marked>

4

BASES (continued)

APPLICABLE SAFETY ANALYSES

INSERT B3.3-118A

Automatic isolation of MFW and main steam line was assumed in the safety analyses to mitigate the consequences of main steam line or MFW line ruptures. The FSAR analyses for steam line breaks (SLBs) was generated before the development and installation of the safety grade EFIC System, which currently performs these automatic safety functions. The FSAR analysis, for example, assumes main steam line isolation through turbine stop valve closure based on an integrated control system signal. This same function is provided by the EFIC System by a safety grade signal that closes the Main Steam Line Isolation valves. The analyses are bounding, and the use of the EFIC System is consistent with the licensing position to take credit for safety grade systems to mitigate the consequences of an accident.

Similarly, vector valve control was not credited in the FSAR SLB analysis. Operator action was credited with isolating EFW to the affected SG within the first 60 seconds. This function would be automatically performed by EFIC. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve logic.

Automatic initiation of EFW is credited in the loss of main feedwater analysis. The automatic actuation was based on the SG low level function of EFIC, although EFIC would initiate EFW based on the loss of both MFW pumps as well.

A The ~~EFIC~~ logic channels satisf~~ies~~y Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 1). 2

LCO

Two channels ~~each~~ of ~~MFW and Main Steam Line Isolation, Vector Valve Enable, and EFW Actuation~~ automatic actuation logic~~s~~ shall be OPERABLE. There are only two channels of automatic actuation logic ~~for Function~~. Therefore, violation of this LCO could result in a complete loss of the automatic Function assuming a single failure of the other channel.

APPLICABILITY A

The ~~MFW and Main Steam Line Isolation~~ automatic actuation logic~~s~~ channels shall be OPERABLE in MODES 1 and 2, and 3 with main steam header pressure ≥ 700 psig because SG inventory can be at a high energy level and can contribute significantly to the peak containment pressure during a

and MODE

(continued)

INSERT B 3.3-118A

MSLB circuitry is installed equipment necessary to automatically isolate main feedwater to both steam generators during a MSLB.

Steam generator outlet pressure is used as input to the MSLB circuitry for detection and feedwater isolation. When a MSLB is sensed, or upon manual actuation, the MFCVs and SFCVs will be closed to isolate the MFW flow paths to both steam generators. In addition, the MFW pumps are tripped. The TDEFW pump will be inhibited from auto-starting or will be auto-stopped if it has already started. A manual override for the TDEFW pump inhibit is provided to allow the operator to subsequently start the TDEFW pump if necessary for heat removal. All of these functions are credited for mitigating a MSLB inside containment.

A-2

5 <Entire Page>

INSERT B3.3-119A

BASES

APPLICABILITY (continued)

secondary side line break. In MODES 4, 5, and 6, the energy level is low and the secondary side feedwater flow rate is low or nonexistent.

The EFW automatic actuation and vector enable logics shall be OPERABLE in MODES 1, 2, and 3 because the SGs are being used for heat removal from the primary system. During these MODES, the core power and heat removal requirements are the greatest, and if the normal source of feedwater is lost, EFW must be initiated rapidly to minimize the overheating of the primary system.

For portions of MODE 4 and for all of MODES 5 and 6, the primary system temperatures are too low to allow the SGs to effectively remove energy.

ACTIONS

If a channel is found inoperable, then all affected logic Functions provided by that channel must be declared inoperable and the LCO Condition entered for the particular protection function affected.

For this LCO, a Note has been added to the ACTIONS indicating that separate Condition entry is allowed for each EFIC logic Function.

A.1

Condition A applies when one or more EFIC logic Functions ~~in~~ a single channel are inoperable (i.e., channel A could be inoperable for all four EFIC logic Functions and Condition A would still be applicable) with all Functions in the other channel OPERABLE. This Condition is equivalent to failure of one EFW, Main Steam Line Isolation, and MFW Isolation train.

With one automatic actuation logic channel inoperable ~~of one or more EFIC Functions~~ inoperable, the associated EFIC train must be restored to OPERABLE status. Since there are only two automatic actuation logic channels per EFIC Function, the condition of one channel inoperable is analogous to having one train of a two train Engineered Safety Feature (ESF) System inoperable. The system safety function can be accomplished; however, a single failure cannot be taken.

INSERT B3.3-119B

(continued)

INSERT B 3.3-119A

Also, during MODE 3, the MFW Isolation function is not required to be OPERABLE when all main feedwater control valves (MFCVs) and startup feedwater control valves (SFCVs) are closed since its function is already fulfilled.

INSERT B 3.3-119B

the channel must be restored to OPERABLE status within 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of MSLB Detection and MFW Isolation Function initiation via individual component controls. The 72 hour Completion Time is consistent with the allowed outage time for the components actuated by the MSLB Detection and MFW Isolation Function.

5 <Except as marked>

A -2

BASES

ACTIONS

A.1 (continued)

Therefore, the failed channel(s) must be restored to OPERABLE status to re-establish the system's single-failure tolerance.

~~Condition A can be thought of as equivalent to failure of a single train of a two train safety system (e.g., the safety function can be accomplished, but a single failure cannot be taken). Thus, the Completion Time of 72 hours has been chosen to be consistent with Completion Times for restoring one inoperable ESF System train.~~

~~The EFIC System has not been analyzed for failure of one train of one Function and the opposite train of the same Function. In this condition, the potential for system interactions that disable heat removal capability on EFW has not been evaluated. Consequently, any combination of failures in both channels A and B is not covered by Condition A and must be addressed by entry into LCO 3.0.3.~~

B.1 and B.2

with both logic channels inoperable or the

~~If~~ Required Action ~~A.1 cannot be met within the required Completion Time~~ and associated Completion Time not met, ~~the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours.~~

INSERT B3.3-120A

The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended functions. This test verifies MFW ~~and Main Steam Line~~ Isolation ~~and EFW initiation~~ automatic actuation logics are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic. ~~The test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of valves associated~~ 4

(continued)

INSERT B 3.3-120A

placed in MODE 3 within 12 hours and the main steam header pressure must be reduced to less than 700 psig or all MFCVs and SFCVs must be closed within 18 hours.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.13.1 (continued)

4 ~~with MFW and Main Steam Line Isolation or actuation of EFW during testing at power.~~ The Frequency of ~~31 days~~ 18 months is based on ~~operating experience, which has demonstrated the rarity of more than one channel failing within the same 31 day interval.~~ INSERT B 3.3-121A

REFERENCES

1. ~~FSAR, Chapter [7].~~ 10 CFR 50.36. 2

INSERT B 3.3-121A

engineering judgment and operating experience that determined testing on an 18 month interval provides reasonable assurance that the circuitry is available to perform its safety function, while the risks of testing during unit operation is avoided.

B 3.3 INSTRUMENTATION

B 3.3.14 Emergency Feedwater Initiation and Control (EFIC) – Emergency Feedwater (EFW) – Vector Valve Logic

BASES

BACKGROUND

The function of the EFW vector valve logic is to determine whether EFW should not be fed to one or the other steam generator. This is to preclude the continued addition of EFW to a depressurized once through steam generator (SG) and, thus, minimize the overcooling effects of a steam leak. Each vector logic may isolate EFW to one SG or the other, never both.

There are four sets of vector valve logic; one in each channel of EFIC. Each set of vector valve logic receives SG pressure information from bistables located in the input logic of the same EFIC channel. The pressure information received is:

a. SG "A" pressure less than 600 psig;

b. SG "B" pressure less than 600 psig;

c. SG "A" pressure 125 psid greater than SG "B" pressure; and

d. SG "B" pressure 125 psid greater than SG "A" pressure.

Each vector valve logic also receives a vector/control enable signal from both EFIC channel A and channel B when EFW is actuated.

The vector valve logic develops signals for open and close control of SG "A" and "B" EFW valves.

The vector valve logic outputs are in a neutral state with the valves fully open until enabled by the control/vector enable from the channel A or B trip logics. When enabled, the vector valve logic can issue close commands to the EFW control valves and open or close commands to the EFW isolation valves per the selected channel assignments.

(continued)

BASES

BACKGROUND (continued)

The valve open/close commands are determined by the relative values of steam generator pressures as follows:

	SG VALVES	
PRESSURE STATUS	"A"	"B"
If SG "A" & SG "B" > 600 psig	Open	Open
If SG "A" > 600 psig & SG "B" < 600 psig	Open	Close
If SG "A" < 600 psig & SG "B" > 600 psig	Close	Open
If SG "A" & SG "B" < 600 psig		
AND		
SG "A" & SG "B" within 125 psid	Open	Open
SG "A" 125 psid > SG "B"	Open	Close
SG "A" 125 psid > SG "A"	Close	Open

APPLICABLE SAFETY ANALYSES

Automatic isolation of main feedwater (MFW) and main steam line was assumed in the safety analyses to mitigate the consequences of main steam line or MFW line ruptures. The FSAR analysis for steam line breaks (SLBs) was generated before the development and installation of the safety grade EFIC System, which currently performs these automatic safety functions. The FSAR analysis, for example, assumes main steam line isolation through turbine stop valve closure based on an integrated control system signal. This same function is provided by the EFIC System by a safety grade signal that closes the main steam line isolation valves. The analyses are bounding, and the use of the EFIC System is consistent with the licensing position to take credit for safety grade systems to mitigate the consequences of an accident.

Similarly, vector logic valve control was not credited in the FSAR SLB analysis. Operator action was credited with isolating EFW to the affected SG within the first

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

60 seconds. This function would be automatically performed by EFIC. Therefore, the FSAR analysis remains conservative relative to the inclusion of the vector valve logic.

EFW vector valve logic response time is included in the required response time for each EFW actuation initiation function instrumentation and is not specified separately.

The EFIC – EFW – vector valve logic satisfies Criterion 3 of the NRC Policy Statement.

LCO

Four channels of the EFIC – EFW – vector valve logic module are required to be OPERABLE. The necessity for four channels is discussed in the BASES for ACTIONS. The 600 psig and 125 psid setpoints were chosen as discussed in Specification B 3.3.11, "EFIC System Instrumentation." The feed only good generator verification study assumed a differential pressure vector value of 150 psid. The 125 psid setpoint conservatively assumes a 25 psi margin for instrument error. Failure to meet this LCO results in not being able to meet the single-failure criterion.

APPLICABILITY

EFIC – EFW – vector valve logic is required in MODES 1, 2, and 3 because the SGs are relied on in these MODES for required RCS heat removal. In MODES 4, 5, and 6, heat removal requirements are reduced and may be provided by the Decay Heat Removal System. Therefore, vector valve logic is not required to be OPERABLE in these MODES.

ACTIONS

A.1

The function of the EFIC-EFW control/isolation valves and the vector valve logic is to meet the single-failure criterion while maintaining the capability to:

a. Provide EFW on demand; and

b. Isolate an SG when required.

(continued)

BASES

ACTIONS

A.1 (continued)

These conflicting requirements result in the necessity for two valves in series, in parallel with two valves in series, and a four channel valve command system.

With one channel inoperable, the system cannot meet the single-failure criterion and still meet the dual functional criteria previously described. Therefore, when one vector valve logic channel is inoperable, the channel must be restored to OPERABLE status within 72 hours. This is analogous to having one EFW train inoperable; wherein a 72 hour Completion Time is provided by the Required Actions of LCO 3.7.4, "EFW System." As such, the Completion Time of 72 hours is based on engineering judgement.

B.1 and B.2

If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

SR 3.3.14.1 is the performance of a CHANNEL FUNCTIONAL TEST every 31 days. This test demonstrates that the EFIC – EFW – vector valve logic performs its function as desired. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

None.

(4)(5) <Entire INSERT>

B 3.3 INSTRUMENTATION

B 3.3.14 Emergency Feedwater (EFW) Pump Initiation Circuitry

BASES

BACKGROUND

EFW pump initiation circuitry is designed to provide safety grade means of controlling the secondary system as a heat sink for core decay heat removal. To ensure the secondary system remains a heat sink, the EFW pump initiation circuitry takes action to initiate EFW when the primary source of feedwater is lost. These actions ensure that a source of cooling water is available to be supplied to a steam generator (SG), thereby establishing the heat sink temperature at the saturation temperature of the secondary system.

EFW is initiated to restore a source of cooling water to the secondary system when conditions indicate that the normal source of feedwater is insufficient to continue heat removal. The EFW pump initiation circuitry contains devices that generate an EFW pump initiation signal when loss of main feedwater pumps are indicated by low hydraulic oil pressure. Each EFW Pump initiation circuit is fed by two loss of main feedwater (LOMF) instrumentation channels (hydraulic oil pressure switches) common only to that circuit which feed a two-out-of-two logic circuit that automatically starts each EFW pump. Each EFW pump also has a dedicated manual start circuit.

Each motor driven EFW pump is normally controlled by a four-position, OFF-AUTO1-AUTO2-RUN, control switch located in the control room. The pump can be manually started by turning the control switch to the RUN position. In the AUTO1 mode, each motor-driven EFW pump starts automatically after a sustained low water level in either steam generator for greater than 30 seconds. In the AUTO2 Mode, each pump starts automatically on low steam generator level or loss of both main feedwater pumps.

The turbine-driven EFW pump is started by opening valve MS-93 which admits steam to the pump turbine. A four-position, RUN-AUTO-OFF-PULL TO LOCK, control switch is provided to control operation of MS-93. The switch is maintained in the AUTO position. In the AUTO mode, MS-93 opens on low hydraulic oil pressure in both MFW pumps.

(continued)

BASES

BACKGROUND (continued)

When the switch is in the RUN position, MS-93 is opened.

Loss of both MFW Pumps was chosen as an EFW automatic initiating parameter because it is a direct and immediate indicator of loss of MFW.

EFW is also initiated by a low level in the SG (after a 30 second delay to prevent spurious actuation) for SG dryout protection. EFW initiation for SG dryout protection is not required by this Specification. Finally, EFW is also initiated by a loss of both MFW pumps as indicated by low hydraulic oil pressure as part of the ATWS Mitigation Circuitry (AMSAC), which is a system provided to comply with the requirements to reduce risk from an anticipated transient without scram (ATWS). EFW initiation for ATWS mitigation is not required by this Specification.

APPLICABLE SAFETY ANALYSES

The transient which forms the basis for initiation of the EFW systems is a loss of MFW transient. In the analysis of the transient, MFW pump turbine low control oil pressure is the parameter assumed to automatically initiate EFW.

The EFW pump initiation circuitry satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two loss of main feedwater (LOMF) pump instrumentation channels and an automatic initiation circuit and a manual initiation circuit are required OPERABLE for each EFW pump. Each LOMF instrumentation channel is considered to include the sensors and measurement channels. The LCO is modified by a Note that limits the OPERABILITY required for the automatic initiation circuitry to MODES 1 and 2.

APPLICABILITY

The initiation circuitry for EFW pumps shall be OPERABLE in MODES 1, 2 and 3 and in MODE 4 when the steam generator is relied upon for heat removal. In MODE 4 when the steam generator is not relied upon for heat removal, and MODES 5, and 6, the primary system temperatures are too low to allow

(continued)

BASES

APPLICABILITY (continued)

the SGs to effectively remove energy and EFW Pump initiation instrumentation is not required to be OPERABLE.

ACTIONS

The ACTIONS are modified by a Note indicating that this Specification may be entered independently for each EFW pump initiation circuit. The Completion Time(s) of the inoperable channels for each EFW automatic initiation circuit are tracked separately for each circuit starting from the time the Condition is entered for that circuit.

A.1

With one or more required EFW pump initiation circuits with one LOMF channel inoperable, the channel(s) must be placed in trip within 1 hour. With the channel in trip, the resultant logic is one-out-of-one. This channel may be considered placed in trip, after tripping, by installing jumpers or by other means that assure the channel remains in the tripped condition.

B.1

With one or more EFW pump initiation circuits inoperable or the Required Action and associated Completion Time of Condition A not met, the affected EFW pump(s) must be declared inoperable immediately since the initiation function is no longer capable of performing its safety function.

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1

A CHANNEL FUNCTIONAL TEST verifies the function of the required trip of the channel. Setpoints for trip must be found within the Allowable Value. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.14.1 (continued)

The Frequency of 31 days is based on operating experience with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.14.2

CHANNEL CALIBRATION is a complete check of the instrument channel including the sensor. The test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channels adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that measurement errors and setpoint errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint analysis.

The Frequency is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Chapters 7 and 15.

2. 10 CFR 50.36.

④⑤ <Entire INSERT>

B 3.3 INSTRUMENTATION

B 3.3.15 Turbine Stop Valve (TSV) Closure

BASES

BACKGROUND

The Turbine Stop Valves (TSV) Closure function partially isolates the main steam lines from the SGs by closing the TSVs on both main steam lines following a high energy line break (HELB).

Two TSVs are provided for each main steam line and are located outside of containment. The TSVs are downstream from the main steam safety valves (MSSVs) and emergency feedwater pump turbine's steam supply to prevent the MSSVs and EFW pump's steam supply from being isolated from the steam generators by TSV closure. Closing the TSVs partially isolates each steam generator from the other, and isolates the turbine from the steam generators.

TSV Closure is initiated by a reactor trip. To keep from rapidly cooling down the primary plant by drawing off too much steam, the turbine is tripped when the reactor trips. Two independent and redundant "Reactor Trip Confirmed" signals in the form of contact closures from the control rod drive system will energize two independent turbine trip mechanisms. The Channel A trip circuit will close all four TSVs within a maximum of 1 second. The Channel B trip circuit will close the TSVs within a maximum of 15 seconds.

APPLICABLE SAFETY ANALYSES

The design basis of the TSV Closure function is established by the analysis for the main steam line break (MSLB) as discussed in the UFSAR, Section 15.13 (Ref. 1). TSV closure is necessary to stop steam flow to the turbine (to prevent overcooling) following all reactor trips.

The accident analysis compares several different MSLB events. The MSLB outside containment upstream of the TSV is limiting for offsite dose, although a break in this section of main steam header has a very low probability. The main MSLB without ICS and without operator action is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available and with a loss of offsite power following turbine trip. With offsite power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System (RCS) cooldown. With a loss of offsite power, the response of mitigating systems, such as the High Pressure Injection (HPI) System pumps, is delayed.

The TSVs remain open during power operation. These valves close upon a reactor trip.

a. For an HELB or an MSLB inside containment, the analysis assumes the TSV in the affected steam generator remains open. For this scenario, steam is discharged into containment from both steam generators until closure of the TSV in the intact steam generator occurs. After TSV closure, steam is discharged into containment only from the affected steam generator.

b. An MSLB outside of containment and upstream from the TSVs is not a containment pressurization concern. The uncontrolled blowdown of both steam generators must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the TSVs isolates the break and limits the blowdown to a single steam generator.

c. An event such as increased steam flow through the turbine will terminate on closing the TSVs.

d. Following a steam generator tube rupture, closure of the TSVs isolates the ruptured steam generator from the intact steam generator.

The TSV Closure function satisfies Criterion 3 of 10 CFR 50.36 (Ref. 2).

LCO

Two TSV Closure channels are required to be OPERABLE.

This LCO provides assurance that the TSVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 limits (Ref. 3).

(continued)

BASES (continued)

APPLICABILITY

Both TSV Closure channels must be OPERABLE in MODES 1, 2 and 3 with any TSVs open. In these conditions when there is significant mass and energy in the RCS and steam generators, the TSV Closure function must be OPERABLE or the TSVs closed. When the TSVs are closed, they are already performing the safety function.

In MODE 4, the steam generator energy is low. Therefore, the TSV Closure channels are not required to be OPERABLE. In MODES 5 and 6, the steam generators do not contain a significant amount of energy because their temperature is below the boiling point of water; therefore, the TSV Closure channels are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

A.1

With one or more TSV Closure channels inoperable, all TSVs must be declared inoperable. A Completion Time of 1 hour is provided to return the TSV Closure channels to OPERABLE status. The 1 hour Completion Time is sufficient time to correct minor problems.

SURVEILLANCE REQUIREMENTS

SR 3.3.15.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST to ensure that the channels can perform their intended function. This test verifies the TSV Closure automatic actuation channels are functional. This test simulates the required inputs to the logic circuit and verifies successful operation of the automatic actuation logic channels. The test need not include actuation of the end device. This is due to the risk of a unit transient caused by the closure of TSVs during testing at power. The Frequency of 31 days is based on engineering judgment and operating experience, which determined the interval provided adequate confidence that the TSV Closure channels are available to perform their safety function, while the risks of testing at operation are avoided.

(continued)

BASES (continued)

REFERENCES	1.	UFSAR, Section 15.13.
	2.	10 CFR 50.36.
	3.	10 CFR 100.

B 3.3 INSTRUMENTATION

B 3.3.~~15~~ 16 2 Reactor Building (RB) Purge Isolation – High Radiation

BASES

BACKGROUND

The RB Purge Isolation – High Radiation Function closes the RB purge valves. This action isolates the RB atmosphere from the environment to minimize releases of radioactivity in the event an accident occurs. ~~The high radiation signal indicates a failure of a barrier to the fuel radioactivity, and most likely a loss of coolant accident. The purge valves must begin to shut rapidly to ensure they reach a completely closed position prior to excessive pressures in the RB, against which the valves may not close.~~ 4

(RIA-45) 2

The radiation monitoring system measures the activity in a representative sample of air drawn in succession through a particulate sampler, an iodine sampler, and a gas sampler. The LCO addresses only the gas sampler portion of this system. ~~The sensitive volume of the gas sampler is shielded with lead and monitored by a Geiger-Mueller detector. The air sample is taken from the center of the purge exhaust duct through an isokinetic nozzle installed in the duct at a point selected for reduced turbulence.~~ 4

~~If a gaseous activity flow rate of approximately 1E-2 Ci/sec (Kr-85) is exceeded, the monitor will alarm and initiate closure of the purge valves. This activity flow rate is selected on the basis of 50,000 scfm flow rate in the purge exhaust and on the basis of a gas monitor setpoint equal to two times the expected background at the location of the monitor, which will provide fast detection of any release. The alarm setpoints for the particulate and iodine channels indicate that an alarm is obtained after the monitor samples a maximum permissible concentration level for 8 hours. Therefore, a maximum of 1.3 mCi of Cs-137 or 67 Ci of DOSE EQUIVALENT I-131 will be released to the atmosphere during this period.~~

21

INSERT B 3.3-126A

The closure of the purge valves ensures the RB remains as a barrier to fission product release. There is no bypass for this function. ~~The closure of the purge valves provides an RB isolation assumed in the accident analysis.~~ 4

(continued)

INSERT B3.3-126A

The trip setpoint is chosen sufficiently below hazardous radiation levels to ensure that the consequences of an accident will be acceptable, provided the unit is operated within the LCOs at the onset of an accident or transient and the equipment functions as designed.

BASES

BACKGROUND (continued)

Trip Setpoints and Allowable Values

21

The trip setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± [rack calibration + comparator setting accuracy]).

The trip setpoints used in the bistables are based on the analytical limits derived from the FSAR, Section [14.1] (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account to allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2). Allowable Values specified in LCO 3.3.15 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the "Unit Specific Setpoint Methodology" (Ref. 3). The actual nominal trip setpoint entered into the bistable is normally still more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

The Allowable Value in SR 3.3.15.3 is based on the methodology described in Reference 3, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

These Allowable Values are established to prevent violation of the accident acceptance criteria during anticipated operational occurrences (AOOs).

Setpoints in accordance with the Allowable Value will ensure that the consequences of Design Basis Accidents (DBAs) will

(continued)

BASES

BACKGROUND

(21)

<u>Trip Setpoints and Allowable Values</u> (continued)

be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed.

APPLICABLE SAFETY ANALYSES

INSERT B 3.3-128A

(4)

The analysis for the maximum hypothetical accident assumes the RB remains intact, with penetrations that are unnecessary for core cooling isolated early in the event, within approximately [60] seconds. The closure of the purge valves ensures the RB integrity assumed in the analysis is maintained. The isolation of the RB has not been analyzed mechanistically in the dose calculations, although its rapid isolation is assumed.

The RB Purge Isolation System satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 2). (2)

LCO

INSERT B 3.3-128B

(5)

For sampling systems, OPERABILITY requires correct valve lineups, sample pump operation, filter motor operation, and detector OPERABILITY, when these sampling features are necessary to initiate a trip as assumed by the safety analysis or setpoint analysis.

Only the Allowable Values are specified for each RB Purge Isolation trip Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by CHANNEL FUNCTIONAL TESTS does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties associated with the trip function. These uncertainties are defined in the "Unit Specific Setpoint Methodology" (Ref. 3) and the Offsite Dose Calculation Manual.

(continued)

INSERT B3.3-128A

During movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). A minimum fuel transfer canal water level and the minimum decay time of 72 hours prior to movement of irradiated fuel assemblies from the reactor ensure that the release of fission product radioactivity subsequent to a fuel handling accident results in doses that are within the guideline values specified in 10 CFR 100. The design basis for fuel handling accidents has historically separated the radiological consequences from the containment capability. The NRC staff has treated the containment capability for fuel handling conditions as a logical part of the "primary success path" to mitigate fuel handling accidents, regardless of the assumptions used to calculate the radiological consequences of such accidents (Ref. 1).

INSERT B3.3-128B

One channel of RB Purge Isolation - High Radiation instrumentation is required to be OPERABLE. OPERABILITY of the instrumentation includes proper operation of the sample pump. This LCO addresses only the gas sampler portion of the System.

BASES

LCO (continued)

[For this unit, the basis for the setpoint Allowable Value is as follows:] 5

APPLICABILITY

instrumentation

The RB purge isolation—high radiation shall be OPERABLE in MODES 1, 2, 3, and 4. Outside of these MODES, the purge isolation must be OPERABLE whenever CORE ALTERATIONS or movement of irradiated fuel assemblies within the RB is taking place. These conditions are those under which the potential for fuel damage, and thus radiation release, is the greatest. While in MODES 5 and 6, without fuel handling in progress, the Purge Valve Isolation System does not need to be OPERABLE because the potential for a radioactive release is minimized and operator action is sufficient to ensure post accident offsite doses are maintained within the limits of 10 CFR 100. The need to use the purge valves in MODES 5 and 6 is in preparation for entry. This capability is required to minimize doses for personnel entering the building and is independent of the automatic isolation capability.

5

INSERT B 3.3-129A

ACTIONS

A.1

5

With one channel inoperable in MODE 1, 2, 3, or 4, the RB purge valves must be placed and maintained in the closed position. This action accomplishes the safety function of the RB Purge Isolation—High Radiation Function. The 1 hour Completion Time is reasonable considering the time required to isolate the penetration and the relative importance of maintaining containment OPERABILITY during MODES 1, 2, 3, and 4.

B.1 and B.2

If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

(continued)

INSERT B3.3-129A

While in MODES 1, 2, 3, and 4, the Purge Valve Isolation System does not need to be OPERABLE because the purge valves are required to be sealed closed.

5 <Except as marked>

BASES

ACTIONS (continued)

A.1, A.2.1, and A.2.2 2
~~C.1, C.2.1, C.2.2~~

Condition ~~C~~ A applies to failure of the high radiation purge function during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB.

With one channel inoperable during CORE ALTERATIONS or during movement of irradiated fuel assemblies within the RB, the RB purge valves must be closed, or CORE ALTERATIONS and movement of irradiated fuel assemblies within the RB must be suspended. Required Action ~~C~~ A.1 accomplishes the function of the high radiation channel. Required Action ~~C~~ A.2.1 and Required Action ~~C~~ A.2.2 place the unit in a configuration in which purge isolation on high radiation is not required. The Completion Time of "Immediately" is consistent with the urgency associated with the loss of RB isolation capability under conditions in which the fuel handling accidents are possible and the high radiation function provides the only automatic actions to mitigate radiation release. 2

SURVEILLANCE REQUIREMENTS

SR 3.3.~~15~~ 16.1 2

SR 3.3.~~15~~ 16.1 is the performance of the CHANNEL CHECK for the RB purge isolation – high radiation instrumentation once every 12 hours to ensure that a gross failure of the instrumentation has not occurred. ~~A~~ CHANNEL CHECK is normally a comparison of the parameter indicated on ~~one~~ the radiation monitoring instrumentation channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. Performance of the CHANNEL CHECK helps to ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared to similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources OPERABLE from the control room, the CHANNEL CHECK should also note the detector's response to these sources. 2

Agreement criteria are ~~determined by the unit staff~~ based on a combination of the channel instrument uncertainties, 2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.~~15~~ 16.1 (continued) 2

including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. The 12 hour Frequency, ~~about once every shift,~~ is based on operating experience that demonstrates channel failure is rare. INSERT B3.3-131A ~~[At this unit, the following administrative controls and design features (e.g., downscale alarms) immediately alert operators to loss of function.]~~ 1

2

SR 3.3.~~15~~ 16.2 2

INSERT B 3.3-131B 5

This SR requires the ~~performance~~ of a CHANNEL FUNCTIONAL TEST ~~once every 92 days~~ to ensure that the channel~~s~~ can perform ~~their~~ its intended function~~s~~. This test verifies the capability of the instrumentation to provide the RB isolation. ~~Any setpoint adjustment shall be consistent with the assumptions of the current unit specific setpoint analysis.~~ 5 2

~~In MODES 1, 2, 3, and 4, the test does not include the actuation of the purge valves, as these valves are normally closed.~~

~~The justification of a 92 day Frequency, in view of the fact that there is only one channel, is Draft NUREG-1366 (Ref. 4).~~ 5

SR 3.3.~~15~~ 16.3 2

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. ~~CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations to ensure that the channel remains operational between successive tests. CHANNEL CALIBRATIONS must be performed consistent with the unit specific setpoint analysis.~~ 5

(continued)

INSERT B3.3-131A

Additionally, control room alarms and annunciators are provided to alert the operator to various "trouble" conditions associated with the instrument.

INSERT B3.3-131B

The frequency requires the isolation capability of the reactor building purge valves to be verified functional once each refueling outage prior to CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. This ensures that this function is verified prior to irradiated fuel assembly handling within containment.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.~~15~~16.3 (continued) (2)

~~The CHANNEL CALIBRATION is a complete check of the instrumentation and detector. In MODES 1, 2, 3, and 4, the CHANNEL CALIBRATION does not include the actuation of the purge valves, since they are normally closed.~~ (5)

The ~~[18]~~ month Frequency is based on ~~the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~ engineering judgment and industry accepted practice. (4)

REFERENCES

~~1. FSAR, Section [14.1].~~

~~2. 10 CFR 50.49.~~ (21)

~~3. "Unit Specific Setpoint Methodology."~~

~~4. Draft NUREG-1366.~~ (5)

2. 10 CFR 50.36. (2)

1. NRC Letter to RG+E dated December 7, 1995, R.E. Ginna Nuclear Power Plant Conversion to Improved Standard Technical Specifications – Resolution of Ginna Design Basis for Refueling Accidents. (4)

B 3.3 INSTRUMENTATION

B 3.3.16 Control Room Isolation – High Radiation

BASES

BACKGROUND

The principal function of the Control Room Isolation – High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. The high radiation isolation function provides assurance that under the required conditions, an isolation signal will be given. The noble gas monitors located in the station vent stack provide isolation and shutdown of the normal Control Room Emergency Ventilation System (CREVS).

The control room isolation signal is provided by a single channel containing an iodine monitor with a scintillation detector and a gaseous monitor with a Geiger-Mueller detector. The iodine channel includes a particulate prefilter with the charcoal cartridge. If a radioactivity concentration above normal background level is detected or if sampling capability is lost, the monitor will initiate a shutdown of the normal duty supply fans and will place the ventilation dampers in their recirculation mode.

<u>Trip Setpoints and Allowable Values</u>

The trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± [rack calibration + comparator setting accuracy]).

The trip setpoints used in the bistables are based on the analytical limits derived from the FSAR, Section [14.1] (Ref. 1). The selection of these trip setpoints indicates that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, and instrument drift, Allowable Values specified in LCO 3.3.15 are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the "Unit Specific Setpoint Methodology" (Ref. 2). The actual nominal trip setpoint

(continued)

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors that are detectable by a CHANNEL FUNCTIONAL TEST. One example of a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

APPLICABLE SAFETY ANALYSES

The CREVS is isolated when a reactor building high pressure Engineered Safety Feature Actuation System signal or a high radiation signal is received. For the first 4 days following a loss of coolant accident, the CREVS is operated in the total recirculation mode. Four days after the start of the accident, the CREVS is started in the intake and recirculation mode and continues to operate in this mode for 30 days. This intake slightly pressurizes the control room. In both cases, the air flows through charcoal filters that are 95% efficient for elemental, particulate, and organic materials. The high radiation function only performs the initial isolation function to begin the recirculation mode of operation.

The Control Room Isolation – High Radiation satisfies Criterion 3 of the NRC Policy Statement.

LCO

Only the Allowable Value is specified for each Control Room Isolation – High Radiation trip Function in the LCO. Nominal trip setpoints are specified in the unit specific setpoint calculations. The nominal setpoints are selected to ensure the setpoint measured by the CHANNEL FUNCTIONAL TEST does not exceed the Allowable Value if the bistable is performing as required. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing is consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than the analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip function. These

(continued)

BASES

LCO (continued)

uncertainties are defined in the "Unit Specific Setpoint Methodology" (Ref. 2).

[At this unit, the basis for the Allowable Value is as follows:]

APPLICABILITY

The control room isolation capability on high radiation shall be OPERABLE whenever there is a chance for an accidental release of radioactivity. This includes MODES 1, 2, 3, 4, [5, and 6] [and during CORE ALTERATIONS] and all MODES and conditions during movement of irradiated fuel assemblies. If a radioactive release were to occur during any of these conditions, the control room would have to remain habitable to ensure reactor shutdown and cooling can be controlled from the main control room.

ACTIONS

A.1

Condition A applies to failure of the Control Room Isolation—High Radiation Function in MODE 1, 2, 3, or 4.

With one channel of Control Room Isolation—High Radiation inoperable, the CREVS must be placed in a condition that does not require the isolation to occur. To ensure that the ventilation system has been placed in a state equivalent to that which occurs after the high radiation isolation has occurred, one OPERABLE train of the CREVS is placed in the emergency recirculation mode of operation. Reactor operation can continue indefinitely in this state. The 1 hour Completion Time is a sufficient amount of time in which to take the Required Action.

The Required Action is modified by a Note, which requires the CREVS be placed in the toxic gas protection mode if automatic transfer to the toxic gas protection mode is inoperable, since the pressurization mode would increase vulnerability to toxic gas releases.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

If the CREVS cannot be placed into recirculation mode while in MODE 1, 2, 3, or 4, actions must be taken to minimize the chances of an accident that could lead to radiation releases. The unit must be placed in at least MODE 3 within 6 hours, with a subsequent cooldown to MODE 5 within 36 hours. This places the reactor in a low energy state that allows greater time for operator action if habitation of the control room is precluded. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

C.1, C.2.1, and C.2.2

Required Action C.1 is the same as discussed earlier for Condition A, except for Completion Time. If the CREVS cannot be placed into recirculation mode during [CORE ALTERATIONS or while] moving irradiated fuel assemblies, then Required Action C.2.1 and Required Action C.2.2 suspend actions that could lead to an accident that could release radioactivity resulting from a fuel handling accident.

Required Action C.2.1 and Required Action C.2.2 place the core in a safe and stable configuration in which it is less likely to experience an accident that could result in a release of radioactivity. The reactor must be maintained in these conditions until the automatic isolation capability is returned to operation or when manual action places one train of the CREVS into the emergency recirculation mode. The Completion Time of "Immediately" for Required Action C.2.1 and Required Action C.2.2 is consistent with the urgency of the situation and accounts for the high radiation function, which provides the only automatic Control Room Isolation Function capable of responding to radiation release due to a fuel handling accident. The Completion Time does not preclude placing any fuel assembly into a safe position before ceasing any such movement.

Note that in certain circumstances, such as fuel handling in the fuel building during power operation, both Condition A and Condition C may apply in the event of channel failure.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.3.16.1

SR 3.3.16.1 is the performance of a CHANNEL CHECK for the Control Room Isolation—High Radiation actuation instrumentation once every 12 hours to ensure that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious.

Performance of the CHANNEL CHECK helps ensure that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation should be compared with similar unit instruments located throughout the unit. If the radiation monitor uses keep alive sources or check sources operated from the control room, the CHANNEL CHECK should also note the detector's response to these sources.

Agreement criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. [At this unit, the following administrative controls and design features (e.g., downscale alarms) immediately alert operators to loss of function.]

SR 3.3.16.2

A Note defines a channel as being OPERABLE for up to 3 hours while bypassed for surveillance testing. The Note allows channel bypass for testing without defining it as inoperable, although during this time period it cannot actuate a control room isolation. This is based on the

(continued)