SUMMARY DESCRIPTION of ITS/BASES CHANGES
ITS SECTION 3.3 - INSTRUMENTATION

TVA is submitting a proposed supplement to TS-362 for ITS Section 3.3, INSTRUMENTATION. This supplement makes several changes associated with NRC comments on Section 3.3, and incorporates changes resulting from internal TVA reviews. A synopsis of the ITS and ITS BASES changes is provided below.

SECTION 3.3.1.1, RPS INSTRUMENTATION

SURVEILLANCE REQUIREMENTS AND CORRESPONDING BASES

In response to TVA reviews, changed frequency on SR 3.3.1.1.7 from 1000 effective full power hours to 1000 MWD/T average core exposure to be consistent with NUREG-1433.

TABLE 3.3.1.1-1 AND CORRESPONDING SAFETY ANALYSIS, LCO, AND APPLICABILITY BASES

In response to TVA reviews, deleted applicability of MODE 5[a] and associated SR FUNCTIONS 2.a, Average Power Range Monitors Neutron Flux - High Setdown, and 2.e, Average Power Range Monitors Neutron Inop. This change is made to be consistent with NUREG-1433.

BASES, BACKGROUND

In response to TVA reviews, made editorial changes to agree with previously submitted NUREG mark-up.

BASES, ACTIONS

In response to an NRC comment, clarified details for Function C.1 to better define when Functions lose trip capability due to multiple inoperable/untripped channels.

SECTION 3.3.2.1, CONTROL ROD BLOCK INSTRUMENTATION

TABLE 3.3.2.1-1 AND CORRESPONDING APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY BASES

In response to TVA reviews, changed footnotes (a) and (b) Minimum Critical Power Ratio (MCPR) values to agree with TS change 353S1 and Amendment No. 249.

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SUMMARY DESCRIPTION of ITS/BASES CHANGES
ITS SECTION 3.3 - INSTRUMENTATION

SECTION 3.3.2.2, FEEDWATER AND MAIN TURBINE HIGH WATER LEVEL TRIP INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

In response to internal TVA reviews, changed Allowable Value for SR 3.3.2.2.3 from 588 inches above vessel zero to 586 inches above vessel zero to agree with change previously made to the NUREG mark-up.

SECTION 3.3.3.1, POST ACCIDENT MONITORING (PAM) INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

In response to an NRC comment, made editorial changes to be consistent with NUREG format.

SECTION 3.3.3.2, BACKUP CONTROL SYSTEM

SURVEILLANCE REQUIREMENTS AND CORRESPONDING BASES

In response to internal TVA reviews, modified SR 3.3.3.2.2 calibration frequency for Unit 3 to agree with engineering analysis.

BASES, TABLE B 3.3.3.2-1

In response to internal TVA reviews, modified Table 3.3.3.2-1 (Backup Control System Instrumentation and Controls) to delete extraneous note from Unit 1 and 2 tables, and deleted from the Unit 3 table the required number of components for parameter 9.

SECTION 3.3.4.1, END OF CYCLE RECIRCULATION PUMP TRIP (EOC-RPT) INSTRUMENTATION

SURVEILLNCE REQUIREMENTS

In response to internal TVA reviews, changed ≥ 10% to ≤ 10% for turbine stop valve closure allowable value to agree with plant design.

SUMMARY DESCRIPTION of ITS/BASES CHANGES
ITS SECTION 3.3 - INSTRUMENTATION

SECTION 3.3.4.2, ANTICIPATED TRANSIENT WITHOUT SCRAM RECIRCULATION PUMP TRIP (ATWS-RPT) INSTRUMENTATION

LCO

In response to an NRC comment, changed water level nomenclature to be consistent other parts of the ITS.

SURVEILLANCE REQUIREMENTS

In response to an NRC comment, modified SR 3.3.4.2.1 water level nomenclature for consistency.

BASES, BACKGROUND, APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

In response to an NRC comment, changed water level nomenclature to be consistent with other parts of the ITS.

SECTION 3.3.5.1, EMERGENCY CORE COOLING SYSTEM (ECCS) INSTRUMENTATION

ACTIONS AND CORRESPONDING BASES

In response to an NRC comment, revised Completion Time descriptive information for Required Actions B.1, C.1, and E.1 to be more consistent with NUREG terminology.

BACKGROUND, APPLICABLE SAFETY ANALYSES, LCO, APPLICABILITY, AND ACTIONS BASES

In response to internal TVA reviews, made editorial changes to agree with changes made in the previously submitted NUREG mark-up and added clarification regarding the effect of the minimum flow bypass valve on High Pressure Coolant Injection System operation.

SUMMARY DESCRIPTION of ITS/BASES CHANGES
ITS SECTION 3.3 - INSTRUMENTATION

SECTION 3.3.6.1, PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

ACTIONS AND CORRESPONDING BASES

In response to an NRC comment, revised the Required Actions. Note A.1 was revised to more clearly delineate the applicability (changed "15 of 16 Channels OPERABLE" to "two or more Channels inoperable"). Required Action D.1 was changed from "Isolate the affected penetration flow path(s)" to original NUREG text, "Isolate associated Main Steam Line (MSL)". Deleted Required Action G.1 and associated Note because Action was redundant to Actions in ITS Section 3.6.1.3, "Primary Containment Isolation Valves".

TABLE 3.3.6.1-1

In response to internal TVA reviews, corrected Functions 2.a and 2.b "Conditions Referenced" from H to G in the NUREG mark-up to agree with changes previously made in identification of Required Actions.

In response to internal TVA reviews, changed Functions 3.a and 4.a "Allowable Value" from 150% rated steam flow for each function to 90 psi for Function 3.a and 450" H_2O for Function 4.a to match corresponding CTS numeric values and supporting analysis.

In response to an NRC comment, revised footnote b to more closely match the NUREG text.

SECTION 3.3.6.2, SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION

SURVEILLANCE REQUIREMENTS

In response to an NRC comment, revised Note 3 to clarify that the 24 hour Allowed Outage Time (AOT) is applicable only for channel calibration or maintenance.

SECTION 3.3.7.1, CONTROL ROOM EMERGENCY VENTILATION (CREV) SYSTEM INSTRUMENTATION

ACTIONS

In response to an NRC comment, revised the Required Actions and Completion Time description for D.1 and D.2 to be consistent with the NUREG.

SUMMARY DESCRIPTION of ITS/BASES CHANGES
ITS SECTION 3.3 - INSTRUMENTATION

SURVEILLANCE REQUIREMENTS AND CORRESPONDING BASES

In response to an NRC comment, revised Note 3 to clarify that the 24 hour AOT is applicable only for channel calibration or maintenance.

SECTION 3.3.8.1, LOSS OF POWER (LOP) INSTRUMENTATION

ACTIONS AND CORRESPONDING BASES

In response to NRC comments, changed the terminology from the relay logic being a channel to a relay being a channel. Revised Required Actions A.1, A.2, B.1, B.2, C.1, C.2, D.1 and D.2 to accommodate the changing of relays to channels.

TABLE 3.3.8.1-1

In response to NRC comments, revised the number of required channels per board for the LOP Functions.

BACKGROUND, APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY BASES

In response to NRC comments, changed the terminology from the relay logic being a channel to a relay being a channel.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 1 ITS LCO SECTION (Revised pages marked *R2)

Replaced 3.3-4 with 3.3-4 Revision 2
Replaced 3.3-6 Revision 1 with 3.3-6 Revision 2
Replaced 3.3-7 Revision 1 with 3.3-7 Revision 2
Replaced 3.3-20 with 3.3-20 Revision 2
Replaced 3.3-22 with 3.3-22 Revision 2
Replaced 3.3-25 Revision 1 with 3.3-25 Revision 2
Replaced 3.3-31 with 3.3-31 Revision 2
Replaced 3.3-32 with 3.3-32 Revision 2
Replaced 3.3-33 with 3.3-33 Revision 2
Replaced 3.3-36 Revision 1 with 3.3-36 Revision 2
Replaced 3.3-37 Revision 1 with 3.3-37 Revision 2
Replaced 3.3-38 Revision 1 with 3.3-38 Revision 2
Replaced 3.3-40 with 3.3.40 Revision 1**
Replaced 3.3-42 through 3.3-47 with 3.3-42 through 3.3-47 Revision 1**
Replaced 3.3-51 Revision 1 with 3.3-51 Revision 2
Replaced 3.3-52 Revision 1 with 3.3-52 Revision 2
Replaced 3.3-53 Revision 1 with 3.3-53 Revision 2
Replaced 3.3-56 Revision 1 with 3.3-56 Revision 2
Replaced 3.3-57 Revision 1 with 3.3-57 Revision 2
Replaced 3.3-58 Revision 1 with 3.3-58 Revision 2
Replaced 3.3-61 Revision 1 with 3.3-61 Revision 2
Replaced 3.3-64 Revision 1 with 3.3-64 Revision 2
Replaced 3.3-65 Revision 1 with 3.3-65 Revision 2
Replaced 3.3-67 Revision 1 with 3.3-67 Revision 2
Replaced 3.3-68 Revision 1 with 3.3-68 Revision 2
Replaced 3.3-69 Revision 1 with 3.3-69 Revision 2
Replaced 3.3-71 Revision 1 with 3.3-71 Revision 2

** These pages inadvertently left out of Supplement 12 submittal

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Functions 1 and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	92 days

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux –High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux –High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 15% RTP
b. Flow Biased Simulated Thermal Power –High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 0.58 W + 62% RTP and ≤ 120% RTP
c. Neutron Flux –High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure –High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level –Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve –Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure –High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level –High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale (Flow Biased)	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(e)
b. Inop	(a),(b)	2	SR 3.3.2.1.1	NA
c. Downscale	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	≥ 3% RTP
2. Rod Worth Minimizer	1(c),2(c)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch –Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(b) THERMAL POWER ≥ 29% and < 90% RTP and MCPR < 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 586 inches above vessel zero.	18 months
SR 3.3.2.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including valve actuation.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1.1	Perform CHANNEL CHECK for each required PAM instrumentation channel.	31 days
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION of the Drywell and Torus H_2 analyzer Functions.	92 days
SR 3.3.3.1.3	Perform CHANNEL CALIBRATION of the Reactor Pressure Functions.	184 days
SR 3.3.3.1.4	Perform CHANNEL CALIBRATION for each required PAM instrumentation channel except for the Reactor Pressure, and the Drywell and Torus H_2 analyzer Functions.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: TSV—Closure: ≤ 10% closed; and TCV Fast Closure, Trip Oil Pressure—Low: ≥ 550 psig.	18 months
SR 3.3.4.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level – Low Low, Level 2; and

b. Reactor Steam Dome Pressure – High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Restore channel to OPERABLE status.	14 days
	OR		
	A.2	--------NOTE-------- Not applicable if inoperable channel is the result of an inoperable breaker. ----------	
		Place channel in trip.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Function with ATWS-RPT trip capability not maintained.	B.1 Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1 Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel Water Level—Low Low, Level 2 Function.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for features in both divisions
	AND		
	B.2	--------NOTE--------- Only applicable for Functions 3.a and 3.b. ---------- Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND		
	B.3	Place channel in trip.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.e, 2.c, 2.d, 1.c, and 2.f. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for features in both divisions
	AND		
	C.2	Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	--------NOTE--------- Only applicable if HPCI pump suction is not aligned to the suppression pool. ---------- Declare HPCI System inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1 --------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Function 1.d. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for subsystems in both divisions
	AND	
	E.2 Restore channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1 Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND	
	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable AND 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported ECCS feature(s) inoperable.	Immediately

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure –High	1,2,3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure –Low (Injection Permissive and ECCS Initiation)	1,2,3	4(b) 2 per trip system	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4 2 per trip system	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Core Spray Pump Discharge Flow –Low (Bypass)	1,2,3, 4(a), 5(a)	2 1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.5	≥ 1647 gpm and ≤ 2910 gpm
e. Core Spray Pump Start –Time Delay Relay					
Pumps A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4 1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Channels affect Common Accident Signal Logic. Refer to LCO 3.8.1, "AC Sources - Operating."

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System (continued)					
e. Core Spray Pump Start -Time Delay Relay (continued)					
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level -Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure -High	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure -Low (Injection Permissive and ECCS Initiation)	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Reactor Steam Dome Pressure -Low (Recirculation Discharge Valve Permissive)	1(c),2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 215 psig and ≤ 245 psig
e. Reactor Vessel Water Level -Level 0	1,2,3	2 1 per subsystem	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 312 5/16 inches above vessel zero

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Deleted.

(c) With associated recirculation pump discharge valve open.

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Low Pressure Coolant Injection Pump Start -Time Delay Relay					
Pump A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	6(e)	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump A (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level -Low Low, Level 2	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(d) With reactor steam dome pressure > 150 psig.

(e) Pumps C and D have 1 relay each and Pumps A and B have 2 relays each (1 per trip system).

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
b. Drywell Pressure –High	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level –High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level –Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level –High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow-Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure –High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level –Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero
e. Core Spray Pump Discharge Pressure –High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure –High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure –High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level –Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 6 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. ADS Trip System B (continued)					
e. Core Spray Pump Discharge Pressure - High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure - High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds

(d) With reactor steam dome pressure > 150 psig.

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 --------NOTE--------- Only applicable for Function 1.d if two or more channels are inoperable. ---------- Place channel in trip.	12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 6.c AND 24 hours for Functions other than Functions 2.a, 2.b, 5.h, 6.b, and 6.c
	AND	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 --------NOTE--------- Only applicable for Function 1.d when 15 of 16 channels are OPERABLE. ---------- Place channel in trip.	30 days
B. One or more Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour OR 4 hours for Function 1.d when normal ventilation is not available
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1 Isolate associated Main Steam Line (MSL).	12 hours
	OR	
	D.2.1 Be in MODE 3.	12 hours
	AND	
	D.2.2 Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time for Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	12 hours 36 hours

(continued)

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure –Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow –High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature –High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level –Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
b. Drywell Pressure –High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow –High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure –Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure –High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

Table 3.3.6.1-1 (page 2 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System Isolation (continued)					
d. HPCI Steam Line Space HPCI Pump Room Area Temperature –High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 200°F
e. HPCI Steam Line Space Torus Area (Exit) Temperature –High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
f. HPCI Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
g. HPCI Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow –High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 450" H_2O
b. RCIC Steam Supply Line Pressure –Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 50 psig
c. RCIC Turbine Exhaust Diaphragm Pressure –High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig
d. RCIC Steam Line Space RCIC Pump Room Area Temperature –High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
e. RCIC Steam Line Space Torus Area (Exit) Temperature –High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F
f. RCIC Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F
g. RCIC Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Main Steam Valve Vault Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 201°F
b. Pipe Trench Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 135°F
c. Pump Room A Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
d. Pump Room B Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
e. Heat Exchanger Room Area (West Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 143°F
f. Heat Exchanger Room Area (East Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
g. SLC System Initiation	1,2	1[a]	H	SR 3.3.6.1.6	NA
h. Reactor Vessel Water Level -Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure -High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 115 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
c. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig

(a) One SLC System Initiation signal provides logic input to close both RWCU valves.

(b) Only one channel per trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.6.2-1 to determine which SRs apply for each Secondary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains secondary containment isolation capability.

3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of a CHANNEL CALIBRATION or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the tripped condition.

SURVEILLANCE		FREQUENCY
SR 3.3.6.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.6.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.2.3	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1	Declare associated CREV subsystem inoperable.	1 hour from discovery of loss of CREV initiation capability
	AND		
	C.2	Place channel in trip.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	D.1	Perform SR 3.3.7.1.2 on the OPERABLE channel.	Once per 24 hours
	AND		
	D.2	Verify alternate monitoring capability.	Once per 12 hours from discovery of both channels inoperable
	AND		
	D.3	Restore one channel to OPERABLE status.	30 days from discovery of both channels inoperable
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1	Place the associated CREV subsystem(s) in the pressurization mode of operation.	1 hour
	OR		
	E.2	Declare associated CREV subsystem inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each CREV Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREV initiation capability.

3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for the performance of a CHANNEL CALIBRATION or maintenance, entry into the associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the trip condition.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	184 days
SR 3.3.7.1.5	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When the associated diesel generator is required to be OPERABLE by LCO 3.8.2, "AC Sources – Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One degraded voltage relay channel inoperable on a shutdown board.	A.1	Verify by administrative means that the other two degraded voltage relay channels and associated timers and the loss of voltage relay channels on that shutdown board are Operable.	Immediately
	AND		
	A.2	Place the degraded voltage relay channel in trip.	15 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more of the loss of voltage relay channels inoperable.	B.1	Verify by administrative means that two or more degraded voltage relay channels and associated timers on that shutdown board are OPERABLE.	Immediately
	AND		
	B.2	Place the inoperable loss of voltage relay channel(s) in trip.	10 days
C. Two or more of the degraded voltage relay channels or one or more associated timer(s) inoperable.	C.1	Verify by administrative means that the loss of voltage relay channels on that shutdown board are OPERABLE.	Immediately
	AND		
	C.2	Place the inoperable degraded voltage relay channel(s) in trip.	10 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more degraded voltage relay channels or one or more associated timers inoperable on one shutdown board. AND The loss of voltage relay channel(s) inoperable on the same shutdown board.	D.1 Verify by administrative means that the other shutdown boards and undervoltage relay channels and associated timers are OPERABLE. AND D.2 Place the inoperable channels in trip.	Immediately 5 days
E. Required Action and associated Completion Time not met.	E.1 Declare associated diesel generator (DG) inoperable.	Immediately

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BOARD	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV Shutdown Board Undervoltage (Loss of Voltage)			
a. Board Undervoltage	2	SR 3.3.8.1.2 SR 3.3.8.1.3	Reset at ≥ 2813 V and ≤ 2927 V
b. Diesel Start Initiation Time Delay	2	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 1.4 seconds and ≤ 1.6 seconds
2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)			
a. Board Undervoltage	3	SR 3.3.8.1.1 SR 3.3.8.1.3	≥ 3900 V and ≤ 3940 V
b.1 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 0.2 seconds and ≤ 0.4 seconds
b.2 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 3 seconds and ≤ 5 seconds
b.3 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 5.15 seconds and ≤ 8.65 seconds
b.4 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 0.9 seconds and ≤ 1.7 seconds

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 2 ITS LCO SECTION (Revised pages marked *R2)

Replaced 3.3-4 with 3.3-4 Revision 2
Replaced 3.3-6 Revision 1 with 3.3-6 Revision 2
Replaced 3.3-7 Revision 1 with 3.3-7 Revision 2
Replaced 3.3-20 with 3.3-20 Revision 2
Replaced 3.3-22 with 3.3-22 Revision 2
Replaced 3.3-25 Revision 1 with 3.3-25 Revision 2
Replaced 3.3-31 with 3.3-31 Revision 2
Replaced 3.3-32 with 3.3-32 Revision 2
Replaced 3.3-33 with 3.3-33 Revision 2
Replaced 3.3-36 with 3.3-36 Revision 2
Replaced 3.3-37 with 3.3-37 Revision 2
Replaced 3.3-38 with 3.3-38 Revision 2
Replaced 3.3-40 with 3.3.40 Revision 1**
Replaced 3.3-42 through 3.3-47 with 3.3-42 through 3.3-47 Revision 1**
Replaced 3.3-51 Revision 1 with 3.3-51 Revision 2
Replaced 3.3-52 Revision 1 with 3.3-52 Revision 2
Replaced 3.3-53 Revision 1 with 3.3-53 Revision 2
Replaced 3.3-56 Revision 1 with 3.3-56 Revision 2
Replaced 3.3-57 Revision 1 with 3.3-57 Revision 2
Replaced 3.3-58 Revision 1 with 3.3-58 Revision 2
Replaced 3.3-61 Revision 1 with 3.3-61 Revision 2
Replaced 3.3-64 Revision 1 with 3.3-64 Revision 2
Replaced 3.3-65 Revision 1 with 3.3-65 Revision 2
Replaced 3.3-67 Revision 1 with 3.3-67 Revision 2
Replaced 3.3-68 Revision 1 with 3.3-68 Revision 2
Replaced 3.3-69 Revision 1 with 3.3-69 Revision 2
Replaced 3.3-71 Revision 1 with 3.3-71 Revision 2

** These pages inadvertently left out of Supplement 12 submittal

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Functions 1 and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	92 days

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux – High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux – High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 15% RTP
b. Flow Biased Simulated Thermal Power – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 0.58 W + 62% RTP and ≤ 120% RTP
c. Neutron Flux – High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure –High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level –Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve –Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure –High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level –High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5[a]	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale (Flow Biased)	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(e)
b. Inop	(a),(b)	2	SR 3.3.2.1.1	NA
c. Downscale	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	≥ 3% RTP
2. Rod Worth Minimizer	1(c),2(c)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch – Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(b) THERMAL POWER ≥ 29% and < 90% RTP and MCPR < 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 586 inches above vessel zero.	18 months
SR 3.3.2.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including valve actuation.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1.1	Perform CHANNEL CHECK for each required PAM instrumentation channel.	31 days
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION of the Drywell and Torus H_2 analyzer Functions.	92 days
SR 3.3.3.1.3	Perform CHANNEL CALIBRATION of the Reactor Pressure Functions.	184 days
SR 3.3.3.1.4	Perform CHANNEL CALIBRATION for each required PAM instrumentation channel except for the Reactor Pressure, and the Drywell and Torus H_2 analyzer Functions.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: TSV – Closure: ≤ 10% closed; and TCV Fast Closure, Trip Oil Pressure – Low: ≥ 550 psig.	18 months
SR 3.3.4.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level – Low Low, Level 2; and

b. Reactor Steam Dome Pressure – High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Restore channel to OPERABLE status.	14 days
	OR		
	A.2	--------NOTE-------- Not applicable if inoperable channel is the result of an inoperable breaker. ----------	
		Place channel in trip.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Function with ATWS-RPT trip capability not maintained.	B.1 Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1 Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel Water Level – Low Low, Level 2 Function.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for features in both divisions
	AND		
	B.2	--------NOTE--------- Only applicable for Functions 3.a and 3.b. ---------- Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND		
	B.3	Place channel in trip.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 1.e, 2.c, 2.d, and 2.f. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for features in both divisions
	AND		
	C.2	Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	--------NOTE--------- Only applicable if HPCI pump suction is not aligned to the suppression pool. ---------- Declare HPCI System inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Function 1.d. ----------	
		Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for subsystems in both divisions
	AND		
	E.2	Restore channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1 Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND	
	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable AND 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported ECCS feature(s) inoperable.	Immediately

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure –High	1,2,3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure –Low (Injection Permissive and ECCS Initiation)	1,2,3	4(b) 2 Per trip system	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4 2 per trip system	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Core Spray Pump Discharge Flow –Low (Bypass)	1,2,3, 4(a), 5(a)	2 1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.5	≥ 1647 gpm and ≤ 2910 gpm
e. Core Spray Pump Start –Time Delay Relay					
Pumps A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4 1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Channels affect Common Accident Signal Logic. Refer to LCO 3.8.1, "AC Sources - Operating."

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System (continued)					
e. Core Spray Pump Start -Time Delay Relay (continued)					
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level -Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure -High	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure -Low (Injection Permissive and ECCS Initiation)	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Reactor Steam Dome Pressure -Low (Recirculation Discharge Valve Permissive)	1(c),2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 215 psig and ≤ 245 psig
e. Reactor Vessel Water Level -Level 0	1,2,3	2 1 per subsystem	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 312 5/16 inches above vessel zero

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Deleted.

(c) With associated recirculation pump discharge valve open.

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Low Pressure Coolant Injection Pump Start -Time Delay Relay					
Pump A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	6(e)	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds
Pump D (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level -Low Low, Level 2	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(d) With reactor steam dome pressure > 150 psig.

(e) Pumps A and B have 1 relay each and Pumps C and D have 2 relays each (1 per trip system).

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
b. Drywell Pressure –High	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level –High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level –Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level –High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow-Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure –High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level -Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero
e. Core Spray Pump Discharge Pressure -High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure -High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level -Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure -High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level -Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 6 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. ADS Trip System B (continued)					
e. Core Spray Pump Discharge Pressure – High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure – High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds

(d) With reactor steam dome pressure > 150 psig.

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 --------NOTE--------- Only applicable for Function 1.d if two or more channels are inoperable. ---------- Place channel in trip.	12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 6.c <u>AND</u> 24 hours for Functions other than Functions 2.a, 2.b, 5.h, 6.b, and 6.c
	<u>AND</u>	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE--------- Only applicable for Function 1.d when 15 of 16 channels are OPERABLE. ---------- Place channel in trip.	30 days
B. One or more Functions with isolation capability not maintained.	B.1	Restore isolation capability.	1 hour OR 4 hours for Function 1.d when normal ventilation is not available
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1	Isolate associated Main Steam Line (MSL).	12 hours
	OR		
	D.2.1	Be in MODE 3.	12 hours
	AND		
	D.2.2	Be in MODE 4.	36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time for Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	12 hours 36 hours

(continued)

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure –Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow –High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature –High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level –Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
b. Drywell Pressure –High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow –High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure –Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure –High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

Table 3.3.6.1-1 (page 2 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System Isolation (continued)					
d. HPCI Steam Line Space HPCI Pump Room Area Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 200°F
e. HPCI Steam Line Space Torus Area (Exit) Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
f. HPCI Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
g. HPCI Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 450" H_2O
b. RCIC Steam Supply Line Pressure – Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 50 psig
c. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig
d. RCIC Steam Line Space RCIC Pump Room Area Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
e. RCIC Steam Line Space Torus Area (Exit) Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F
f. RCIC Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F
g. RCIC Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Main Steam Valve Vault Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 201°F
b. Pipe Trench Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 135°F
c. Pump Room A Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
d. Pump Room B Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
e. Heat Exchanger Room Area (West Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 143°F
f. Heat Exchanger Room Area (East Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
g. SLC System Initiation	1,2	1[a]	H	SR 3.3.6.1.6	NA
h. Reactor Vessel Water Level -Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure -High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 115 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
c. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig

(a) One SLC System Initiation signal provides logic input to close both RWCU valves.

(b) Only one channel per trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.6.2-1 to determine which SRs apply for each Secondary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains secondary containment isolation capability.

3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of a CHANNEL CALIBRATION or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the tripped condition.

SURVEILLANCE		FREQUENCY
SR 3.3.6.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.6.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.2.3	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1	Declare associated CREV subsystem inoperable.	1 hour from discovery of loss of CREV initiation capability
	AND		
	C.2	Place channel in trip.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	D.1	Perform SR 3.3.7.1.2 on the OPERABLE channel.	Once per 24 hours
	AND		
	D.2	Verify alternate monitoring capability.	Once per 12 hours from discovery of both channels inoperable
	AND		
	D.3	Restore one channel to OPERABLE status.	30 days from discovery of both channels inoperable
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1	Place the associated CREV subsystem(s) in the pressurization mode of operation.	1 hour
	OR		
	E.2	Declare associated CREV subsystem inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each CREV Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREV initiation capability.

3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for the performance of a CHANNEL CALIBRATION or maintenance, entry into the associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the trip condition.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	184 days
SR 3.3.7.1.5	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When the associated diesel generator is required to be OPERABLE by LCO 3.8.2, "AC Sources—Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One degraded voltage relay channel inoperable on a shutdown board.	A.1	Verify by administrative means that the other two degraded voltage relay channels and associated timers and the loss of voltage relay channels on that shutdown board are Operable.	Immediately
	AND		
	A.2	Place the degraded voltage relay channel in trip.	15 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more of the loss of voltage relay channels inoperable.	B.1	Verify by administrative means that two or more degraded voltage relay channels and associated timers on that shutdown board are OPERABLE.	Immediately
	AND		
	B.2	Place the inoperable loss of voltage relay channel(s) in trip.	10 days
C. Two or more of the degraded voltage relay channels or one or more associated timer(s) inoperable.	C.1	Verify by administrative means that the loss of voltage relay channels on that shutdown board are OPERABLE.	Immediately
	AND		
	C.2	Place the inoperable degraded voltage relay channel(s) in trip.	10 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more degraded voltage relay channels or one or more associated timers inoperable on one shutdown board. AND The loss of voltage relay channel(s) inoperable on the same shutdown board.	D.1 Verify by administrative means that the other shutdown boards and undervoltage relay channels and associated timers are OPERABLE. AND D.2 Place the inoperable channels in trip.	Immediately 5 days
E. Required Action and associated Completion Time not met.	E.1 Declare associated diesel generator (DG) inoperable.	Immediately

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BOARD	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV Shutdown Board Undervoltage (Loss of Voltage)			
a. Board Undervoltage	2	SR 3.3.8.1.2 SR 3.3.8.1.3	Reset at ≥ 2813 V and ≤ 2927 V
b. Diesel Start Initiation Time Delay	2	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 1.4 seconds and ≤ 1.6 seconds
2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)			
a. Board Undervoltage	3	SR 3.3.8.1.1 SR 3.3.8.1.3	≥ 3900 V and ≤ 3940 V
b.1 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 0.2 seconds and ≤ 0.4 seconds
b.2 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 3 seconds and ≤ 5 seconds
b.3 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 5.15 seconds and ≤ 8.65 seconds
b.4 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 0.9 seconds and ≤ 1.7 seconds

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 3 ITS LCO SECTION (Revised pages marked *R2)

Replaced 3.3-4 with 3.3-4 Revision 2
Replaced 3.3-6 Revision 1 with 3.3-6 Revision 2
Replaced 3.3-7 Revision 1 with 3.3-7 Revision 2
Replaced 3.3-20 with 3.3-20 Revision 2
Replaced 3.3-22 with 3.3-22 Revision 2
Replaced 3.3-25 Revision 1 with 3.3-25 Revision 2
Replaced 3.3-28 Revision 1 with 3.3-28 Revision 2
Replaced 3.3-31 with 3.3-31 Revision 2
Replaced 3.3-32 with 3.3-32 Revision 2
Replaced 3.3-33 with 3.3-33 Revision 2
Replaced 3.3-36 with 3.3-36 Revision 2
Replaced 3.3-37 with 3.3-37 Revision 2
Replaced 3.3-38 with 3.3-38 Revision 2
Replaced 3.3-40 with 3.3.40 Revision 1**
Replaced 3.3-42 through 3.3-47 with 3.3-42 through 3.3-47 Revision 1**
Replaced 3.3-51 Revision 1 with 3.3-51 Revision 2
Replaced 3.3-52 Revision 1 with 3.3-52 Revision 2
Replaced 3.3-53 Revision 1 with 3.3-53 Revision 2
Replaced 3.3-56 Revision 1 with 3.3-56 Revision 2
Replaced 3.3-57 Revision 1 with 3.3-57 Revision 2
Replaced 3.3-58 Revision 1 with 3.3-58 Revision 2
Replaced 3.3-61 Revision 1 with 3.3-61 Revision 2
Replaced 3.3-64 Revision 1 with 3.3-64 Revision 2
Replaced 3.3-65 Revision 1 with 3.3-65 Revision 2
Replaced 3.3-67 Revision 1 with 3.3-67 Revision 2
Replaced 3.3-68 Revision 1 with 3.3-68 Revision 2
Replaced 3.3-69 Revision 1 with 3.3-69 Revision 2
Replaced 3.3-71 Revision 1 with 3.3-71 Revision 2

** These pages inadvertently left out of Supplement 12 submittal

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.4	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.6	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.7	Calibrate the local power range monitors.	1000 MWD/T average core exposure
SR 3.3.1.1.8	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.1.1.9	----------NOTES---------- 1. Neutron detectors are excluded. 2. For Functions 1 and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	92 days

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux –High	2	3	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
	5[a]	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120/125 divisions of full scale
b. Inop	2	3	G	SR 3.3.1.1.3 SR 3.3.1.1.14	NA
	5[a]	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux –High, Setdown	2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.3 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 15% RTP
b. Flow Biased Simulated Thermal Power –High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 0.58 W + 62% RTP and ≤ 120% RTP
c. Neutron Flux –High	1	2	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.14	≤ 120% RTP

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
d. Downscale	1	2	F	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	≥ 3% RTP
e. Inop	1,2	2	G	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	NA
3. Reactor Vessel Steam Dome Pressure –High	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.10 SR 3.3.1.1.14	≤ 1055 psig
4. Reactor Vessel Water Level –Low, Level 3	1,2	2	G	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≥ 538 inches above vessel zero
5. Main Steam Isolation Valve –Closure	1	8	F	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 10% closed
6. Drywell Pressure –High	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 2.5 psig
7. Scram Discharge Volume Water Level –High					
a. Resistance Temperature Detector	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
b. Float Switch	1,2	2	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons
	5(a)	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 50 gallons

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Upscale (Flow Biased)	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	(e)
b. Inop	(a),(b)	2	SR 3.3.2.1.1	NA
c. Downscale	(a),(b)	2	SR 3.3.2.1.1 SR 3.3.2.1.4	≥ 3% RTP
2. Rod Worth Minimizer	1(c),2(c)	1	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.7	NA
3. Reactor Mode Switch —Shutdown Position	(d)	2	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ 90% RTP and MCPR < 1.44.

(b) THERMAL POWER ≥ 29% and < 90% RTP and MCPR < 1.75.

(c) With THERMAL POWER ≤ 10% RTP.

(d) Reactor mode switch in the shutdown position.

(e) Less than or equal to the Allowable Value specified in the COLR.

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.2.2.3	Perform CHANNEL CALIBRATION. The Allowable Value shall be ≤ 586 inches above vessel zero.	18 months
SR 3.3.2.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including valve actuation.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1.1	Perform CHANNEL CHECK for each required PAM instrumentation channel.	31 days
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION of the Drywell and Torus H_2 analyzer Functions.	92 days
SR 3.3.3.1.3	Perform CHANNEL CALIBRATION of the Reactor Pressure Functions.	184 days
SR 3.3.3.1.4	Perform CHANNEL CALIBRATION for each required PAM instrumentation channel except for the Reactor Pressure, and the Drywell and Torus H_2 analyzer Functions.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.2.1	Verify each required control circuit and transfer switch is capable of performing the intended function.	18 months
SR 3.3.3.2.2	Perform CHANNEL CALIBRATION for the Suppression Pool Water Level Function.	18 months
SR 3.3.3.2.3	Perform CHANNEL CALIBRATION for each required instrumentation channel except for the Suppression Pool Water Level Function.	18 months

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: TSV – Closure: ≤ 10% closed; and TCV Fast Closure, Trip Oil Pressure – Low: ≥ 550 psig.	18 months
SR 3.3.4.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	18 months

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level—Low Low, Level 2; and

b. Reactor Steam Dome Pressure—High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Restore channel to OPERABLE status.	14 days
	OR		
	A.2	--------NOTE-------- Not applicable if inoperable channel is the result of an inoperable breaker. ----------	
		Place channel in trip.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Function with ATWS-RPT trip capability not maintained.	B.1 Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1 Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel Water Level – Low Low, Level 2 Function.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, and 2.b. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for features in both divisions
	AND		
	B.2	--------NOTE--------- Only applicable for Functions 3.a and 3.b. ---------- Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND		
	B.3	Place channel in trip.	24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c, 1.e, 2.c, 2.d, and 2.f. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for features in both divisions
	AND		
	C.2	Restore channel to OPERABLE status.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1	--------NOTE--------- Only applicable if HPCI pump suction is not aligned to the suppression pool. ---------- Declare HPCI System inoperable.	1 hour

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1 --------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Function 1.d. ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of initiation capability for subsystems in both divisions
	AND	
	E.2 Restore channel to OPERABLE status.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	G.1 Declare ADS valves inoperable.	1 hour from discovery of loss of ADS initiation capability in both trip systems
	AND	
	G.2 Restore channel to OPERABLE status.	96 hours from discovery of inoperable channel concurrent with HPCI or RCIC inoperable AND 8 days
H. Required Action and associated Completion Time of Condition B, C, D, E, F, or G not met.	H.1 Declare associated supported ECCS feature(s) inoperable.	Immediately

Table 3.3.5.1-1 (page 1 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4(b)	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure – High	1,2,3	4(b)	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure – Low (Injection Permissive and ECCS Initiation)	1,2,3	4(b) 2 per trip system	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4 2 per trip system	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Core Spray Pump Discharge Flow – Low (Bypass)	1,2,3, 4(a), 5(a)	2 1 per subsystem	E	SR 3.3.5.1.2 SR 3.3.5.1.5	≥ 1647 gpm and ≤ 2910 gpm
e. Core Spray Pump Start – Time Delay Relay					
Pumps A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	4 1 per pump	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump A (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Channels affect Common Accident Signal Logic. Refer to LCO 3.8.1, "AC Sources - Operating."

Table 3.3.5.1-1 (page 2 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Core Spray System (continued)					
e. Core Spray Pump Start -Time Delay Relay (continued)					
Pump D (with normal power)	1,2,3, 4(a), 5(a)	1	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
2. Low Pressure Coolant Injection (LPCI) System					
a. Reactor Vessel Water Level -Low Low Low, Level 1	1,2,3, 4(a), 5(a)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure -High	1,2,3	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Steam Dome Pressure -Low (Injection Permissive and ECCS Initiation)	1,2,3	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
	4(a), 5(a)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 435 psig and ≤ 465 psig
d. Reactor Steam Dome Pressure -Low (Recirculation Discharge Valve Permissive)	1(c),2(c), 3(c)	4	C	SR 3.3.5.1.2 SR 3.3.5.1.4 SR 3.3.5.1.6	≥ 215 psig and ≤ 245 psig
e. Reactor Vessel Water Level -Level 0	1,2,3	2 1 per subsystem	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 312 5/16 inches above vessel zero

(continued)

(a) When associated subsystem(s) are required to be OPERABLE.

(b) Deleted.

(c) With associated recirculation pump discharge valve open.

Table 3.3.5.1-1 (page 3 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. LPCI System (continued)					
f. Low Pressure Coolant Injection Pump Start -Time Delay Relay					
Pump A,B,C,D (with diesel power)	1,2,3, 4(a), 5(a)	8(e)	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump A (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 0 seconds and ≤ 1 second
Pump B (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 6 seconds and ≤ 8 seconds
Pump C (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 12 seconds and ≤ 16 seconds
Pump D (with normal power)	1,2,3, 4(a), 5(a)	2 1 per trip system	C	SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 18 seconds and ≤ 24 seconds
3. High Pressure Coolant Injection (HPCI) System					
a. Reactor Vessel Water Level -Low Low, Level 2	1, 2(d), 3(d)	4	B	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 470 inches above vessel zero

(continued)

(a) When the associated subsystem(s) are required to be OPERABLE.

(d) With reactor steam dome pressure > 150 psig.

(e) Pumps A, B, C, and D have 2 relays each (1 per trip system).

Table 3.3.5.1-1 (page 4 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System (continued)					
b. Drywell Pressure -High	1, 2(d), 3(d)	4	B	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Reactor Vessel Water Level -High, Level 8	1, 2(d), 3(d)	2	C	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 583 inches above vessel zero
d. Condensate Header Level -Low	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ Elev. 551 feet
e. Suppression Pool Water Level -High	1, 2(d), 3(d)	1	D	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≤ 7 inches above instrument zero
f. High Pressure Coolant Injection Pump Discharge Flow-Low (Bypass)	1, 2(d), 3(d)	1	E	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 671 gpm
4. Automatic Depressurization System (ADS) Trip System A					
a. Reactor Vessel Water Level -Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure -High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 5 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. ADS Trip System A (continued)					
d. Reactor Vessel Water Level –Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero
e. Core Spray Pump Discharge Pressure –High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure –High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds
5. ADS Trip System B					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1, 2(d), 3(d)	2	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 398 inches above vessel zero
b. Drywell Pressure –High	1, 2(d), 3(d)	2	F	SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 2.5 psig
c. Automatic Depressurization System Initiation Timer	1, 2(d), 3(d)	1	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 115 seconds
d. Reactor Vessel Water Level –Low, Level 3 (Confirmatory)	1, 2(d), 3(d)	1	F	SR 3.3.5.1.1 SR 3.3.5.1.2 SR 3.3.5.1.5 SR 3.3.5.1.6	≥ 544 inches above vessel zero

(continued)

(d) With reactor steam dome pressure > 150 psig.

Table 3.3.5.1-1 (page 6 of 6)
Emergency Core Cooling System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER FUNCTION	CONDITIONS REFERENCED FROM REQUIRED ACTION A.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. ADS Trip System B (continued)					
e. Core Spray Pump Discharge Pressure –High	1, 2(d), 3(d)	4	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 175 psig and ≤ 195 psig
f. Low Pressure Coolant Injection Pump Discharge Pressure –High	1, 2(d), 3(d)	8	G	SR 3.3.5.1.2 SR 3.3.5.1.3 SR 3.3.5.1.6	≥ 90 psig and ≤ 110 psig
g. Automatic Depressurization System High Drywell Pressure Bypass Timer	1, 2(d), 3(d)	2	G	SR 3.3.5.1.5 SR 3.3.5.1.6	≤ 322 seconds

(d) With reactor steam dome pressure > 150 psig.

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 --------NOTE--------- Only applicable for Function 1.d if two or more channels are inoperable. ---------- Place channel in trip.	12 hours for Functions 2.a, 2.b, 5.h, 6.b, and 6.c AND 24 hours for Functions other than Functions 2.a, 2.b, 5.h, 6.b, and 6.c
	AND	(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	--------NOTE--------- Only applicable for Function 1.d when 15 of 16 channels are OPERABLE. ---------- Place channel in trip.	30 days
B. One or more Functions with isolation capability not maintained.	B.1	Restore isolation capability.	1 hour OR 4 hours for Function 1.d when normal ventilation is not available
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1	Isolate associated Main Steam Line (MSL).	12 hours
	OR		
	D.2.1	Be in MODE 3.	12 hours
	AND		
	D.2.2	Be in MODE 4.	36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1 Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1 Isolate the affected penetration flow path(s).	1 hour
G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. OR Required Action and associated Completion Time for Condition F not met.	G.1 Be in MODE 3. AND G.2 Be in MODE 4.	12 hours 36 hours

(continued)

Table 3.3.6.1-1 (page 1 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level –Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 398 inches above vessel zero
b. Main Steam Line Pressure –Low	1	2	E	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 825 psig
c. Main Steam Line Flow –High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 140% rated steam flow
d. Main Steam Tunnel Temperature –High	1,2,3	8	D	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 200°F
2. Primary Containment Isolation					
a. Reactor Vessel Water Level –Low, Level 3	1,2,3	2	G	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
b. Drywell Pressure –High	1,2,3	2	G	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow –High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 90 psi
b. HPCI Steam Supply Line Pressure –Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 100 psig
c. HPCI Turbine Exhaust Diaphragm Pressure –High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig

(continued)

Table 3.3.6.1-1 (page 2 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. HPCI System Isolation (continued)					
d. HPCI Steam Line Space HPCI Pump Room Area Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 200°F
e. HPCI Steam Line Space Torus Area (Exit) Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
f. HPCI Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
g. HPCI Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
4. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 450" H_2O
b. RCIC Steam Supply Line Pressure – Low	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 50 psig
c. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	3	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 20 psig
d. RCIC Steam Line Space RCIC Pump Room Area Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 180°F
e. RCIC Steam Line Space Torus Area (Exit) Temperature – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F
f. RCIC Steam Line Space Torus Area (Midway) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F
g. RCIC Steam Line Space Torus Area (Entry) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.6	≤ 155°F

(continued)

Table 3.3.6.1-1 (page 3 of 3)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. Reactor Water Cleanup (RWCU) System Isolation					
a. Main Steam Valve Vault Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 201°F
b. Pipe Trench Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 135°F
c. Pump Room A Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
d. Pump Room B Area Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 152°F
e. Heat Exchanger Room Area (West Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 143°F
f. Heat Exchanger Room Area (East Wall) Temperature - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.6	≤ 170°F
g. SLC System Initiation	1,2	1[a]	H	SR 3.3.6.1.6	NA
h. Reactor Vessel Water Level -Low, Level 3	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
6. Shutdown Cooling System Isolation					
a. Reactor Steam Dome Pressure -High	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 115 psig
b. Reactor Vessel Water Level - Low, Level 3	3,4,5	2[b]	I	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 538 inches above vessel zero
c. Drywell Pressure - High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig

(a) One SLC System Initiation signal provides logic input to close both RWCU valves.

(b) Only one channel per trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.6.2-1 to determine which SRs apply for each Secondary Containment Isolation Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains secondary containment isolation capability.

3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of a CHANNEL CALIBRATION or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the tripped condition.

SURVEILLANCE		FREQUENCY
SR 3.3.6.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.6.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.2.3	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.2.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	C.1	Declare associated CREV subsystem inoperable.	1 hour from discovery of loss of CREV initiation capability
	AND		
	C.2	Place channel in trip.	24 hours
D. As required by Required Action A.1 and referenced in Table 3.3.7.1-1.	D.1	Perform SR 3.3.7.1.2 on the OPERABLE channel.	Once per 24 hours
	AND		
	D.2	Verify alternate monitoring capability.	Once per 12 hours from discovery of both channels inoperable
	AND		
	D.3	Restore one channel to OPERABLE status.	30 days from discovery of both channels inoperable
E. Required Action and associated Completion Time of Condition B, C, or D not met.	E.1	Place the associated CREV subsystem(s) in the pressurization mode of operation.	1 hour
	OR		
	E.2	Declare associated CREV subsystem inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

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

1. Refer to Table 3.3.7.1-1 to determine which SRs apply for each CREV Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains CREV initiation capability.

3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for the performance of a CHANNEL CALIBRATION or maintenance, entry into the associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the trip condition.

SURVEILLANCE		FREQUENCY
SR 3.3.7.1.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.7.1.2	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.7.1.3	Perform CHANNEL CALIBRATION.	92 days
SR 3.3.7.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST.	184 days
SR 3.3.7.1.5	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.7.1.6	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.8.1 Loss of Power (LOP) Instrumentation

LCO 3.3.8.1 The LOP instrumentation for each Function in Table 3.3.8.1-1 shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
When the associated diesel generator is required to be OPERABLE by LCO 3.8.2, "AC Sources – Shutdown."

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One degraded voltage relay channel inoperable on a shutdown board.	A.1	Verify by administrative means that the other two degraded voltage relay channels and associated timers and the loss of voltage relay channels on that shutdown board are Operable.	Immediately
	AND		
	A.2	Place the degraded voltage relay channel in trip.	15 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One or more of the loss of voltage relay channels inoperable.	B.1	Verify by administrative means that two or more degraded voltage relay channels and associated timers on that shutdown board are OPERABLE.	Immediately
	AND		
	B.2	Place the inoperable loss of voltage relay channel(s) in trip.	10 days
C. Two or more of the degraded voltage relay channels or one or more associated timer(s) inoperable.	C.1	Verify by administrative means that the loss of voltage relay channels on that shutdown board are OPERABLE.	Immediately
	AND		
	C.2	Place the inoperable degraded voltage relay channel(s) in trip.	10 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Two or more degraded voltage relay channels or one or more associated timers inoperable on one shutdown board. AND The loss of voltage relay channel(s) inoperable on the same shutdown board.	D.1 Verify by administrative means that the other shutdown boards and undervoltage relay channels and associated timers are OPERABLE. AND D.2 Place the inoperable channels in trip.	Immediately 5 days
E. Required Action and associated Completion Time not met.	E.1 Declare associated diesel generator (DG) inoperable.	Immediately

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER BOARD	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV Shutdown Board Undervoltage (Loss of Voltage)			
a. Board Undervoltage	2	SR 3.3.8.1.2 SR 3.3.8.1.3	Reset at ≥ 2813 V and ≤ 2927 V
b. Diesel Start Initiation Time Delay	2	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 1.4 seconds and ≤ 1.6 seconds
2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)			
a. Board Undervoltage	3	SR 3.3.8.1.1 SR 3.3.8.1.3	≥ 3900 V and ≤ 3940 V
b.1 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 0.2 seconds and ≤ 0.4 seconds
b.2 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 3 seconds and ≤ 5 seconds
b.3 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 5.15 seconds and ≤ 8.65 seconds
b.4 Time Delay	1	SR 3.3.8.1.2 SR 3.3.8.1.3	≥ 0.9 seconds and ≤ 1.7 seconds

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 1 ITS BASES (Revised pages marked *R2)

Replaced B 3.3-1 with B 3.3-1 Revision 2
Replaced B 3.3-8 Revision 1 with B 3.3-8 Revision 2
Replaced B 3.3-24 Revision 1 with B 3.3-24 Revision 2
Replaced B 3.3-25 Revision 1 with B 3.3-25 Revision 2
Replaced B 3.3-30 Revision 1 with B 3.3-30 Revision 2
Replaced B 3.3-31 Revision 1 with B 3.3-31 Revision 2
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Replaced B 3.3-80 Revision 1 with B 3.3-80 Revision 2
Replaced B 3.3-92 Revision 1 with B 3.3-92 Revision 2
Replaced B 3.3-94 Revision 1 with B 3.3-94 Revision 2
Replaced B 3.3-95 Revision 1 with B 3.3-95 Revision 2
Replaced B 3.3-101 Revision 1 with B 3.3-101 Revision 2
Replaced B 3.3-102 Revision 1 with B 3.3-102 Revision 2
Replaced B 3.3-104 Revision 1 with B 3.3-104 Revision 2
Replaced B 3.3-105 Revision 1 with B 3.3-105 Revision 2
Replaced B 3.3-106 Revision 1 with B 3.3-106 Revision 2
Replaced B 3.3-108 Revision 1 with B 3.3-108 Revision 2
Replaced B 3.3-109 Revision 1 with B 3.3-109 Revision 2
Replaced B 3.3-125 Revision 1 with B 3.3-125 Revision 2
Replaced B 3.3-128 Revision 1 with B 3.3-128 Revision 2
Replaced B 3.3-129 Revision 1 with B 3.3-129 Revision 2
Replaced B 3.3-130 Revision 1 with B 3.3-130 Revision 2
Replaced B 3.3-168 Revision 1 with B 3.3-168 Revision 2
Replaced B 3.3-169 Revision 1 with B 3.3-169 Revision 2
Replaced B 3.3-170 Revision 1 with B 3.3-170 Revision 2
Replaced B 3.3-171 Revision 1 with B 3.3-171 Revision 2
Replaced B 3.3-172 Revision 1 with B 3.3-172 Revision 2
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Replaced B 3.3-196 Revision 1 with B 3.3-196 Revision 2
Replaced B 3.3-197 Revision 1 with B 3.3-197 Revision 2
Replaced B 3.3-200 Revision 1 with B 3.3-200 Revision 2
Replaced B 3.3-202 Revision 1 with B 3.3-202 Revision 2
Replaced B 3.3-203 Revision 1 with B 3.3-203 Revision 2
Replaced B 3.3-204 Revision 1 with B 3.3-204 Revision 2
Replaced B 3.3-205 Revision 1 with B 3.3-205 Revision 2
Replaced B 3.3-206 Revision 1 with B 3.3-206 Revision 2
Replaced B 3.3-207 Revision 1 with B 3.3-207 Revision 2
Replaced B 3.3-208 Revision 1 with B 3.3-208 Revision 2

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure 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 LCOs on other reactor system parameters and equipment performance. The LSSS are defined in this Specification as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).

The RPS, as described in the FSAR, Section 7.2 (Ref. 1), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure trip oil pressure (indicated by TCV low hydraulic pressure), turbine stop valve (TSV) position, drywell pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position, manual, and RPS channel test switch scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown, manual and RPS channel test switch scram signals). Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay deenergizes, which then outputs an RPS trip signal to the trip logic.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux - High, Setdown (continued)

14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

decision of which trip system is in the more degraded state should be based on prudent judgment and take into account current plant conditions (i.e., what MODE the plant is in). If this action would result in a scram or RPT, it is permissible to place the other trip system or its inoperable channels in trip.

The 6 hour Completion Time is judged acceptable based on the remaining capability to trip, the diversity of the sensors available to provide the trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of a scram.

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve - Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).

(continued)

BASES

ACTIONS

C.1 (continued)

For Function 8 (Turbine Stop Valve - Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip).

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A, B, or C and the associated Completion Time has expired, Condition D will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

E.1, F.1, and G.1

If the channel(s) is not restored to OPERABLE status or placed in trip (or the associated trip system placed in trip) within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. The allowed Completion Times are reasonable, based on operating experience, to reach the specified condition from full power conditions in an orderly manner and without challenging plant systems. In addition,

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.5 and SR 3.3.1.1.6 (continued)

If overlap for a group of channels is not demonstrated (e.g., IRM/APRM overlap), the reason for the failure of the Surveillance should be determined and the appropriate channel(s) declared inoperable. Only those appropriate channels that are required in the current MODE or condition should be declared inoperable.

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/T average core exposure Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8 and SR 3.3.1.1.12

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.9, SR 3.3.1.1.10, and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This 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 drifts between successive calibrations consistent with the plant specific setpoint methodology. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 includes physical inspection and actuation of the switches.

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from 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 performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 29% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.75, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.44, no RWE event will result in exceeding the MCPR

(continued)

Table B 3.3.3.2-1 (Page 1 of 3)
Backup Control System Instrumentation and Controls

FUNCTION	NUMBER REQUIRED
Instrument Parameter	
1. Reactor Water Level Indication	1
2. Reactor Pressure Indication	1
3. Suppression Pool Temperature Indication	1
4. Suppression Pool Level Indication	1
5. Drywell Pressure Indication	1
6. RHR Flow Indication	1
7. RCIC Flow Indication	1, note a
8. RCIC Turbine Speed Indication	1
9. Drywell Temperature Indication	1
10. RHRSW Header Pressure	1, note p
Transfer/Control Parameter	
11. Main Steam Relief Valve (MSRV) Transfer & Control	3, note b
12. Main Steam Isolation Valve (MSIV) Transfer & Control (Closure)	4, note c
13. Main Steam Drain Line Isolation Valve	1, note d
14. RHRSW Pumps	note e
15. RHRSW Discharge Valves for RHR Loop I Heat Exchangers	2, note f
note a: RCIC flow indication may be obtained from the Flow Indicating Controller	
note b: 1 required for each of 3 MSRVs.	
note c: 1 MSIV required per penetration, may be either inboard valve or outboard valve.	
note d: 1 Main Steam Drain Line isolation valve required, may be either inboard valve or outboard valve.	
note e: There are 12 RHRSW pumps. All are equipped with emergency transfer switches. 2 of the 12 must be available for EECW service (supports all units) and an additional 1 must be available for RHRSW service.	
note f: 1 Discharge Valve per RHR Loop I Heat Exchanger for a total of 2.	
note o: Note not used.	
note p: The RHRSW Pressure indicator for the Header of the RHRSW Pump that supports RHR service is required.	

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level - Low Low, Level 2 or Reactor Steam Dome Pressure - High setpoint is reached, the recirculation pump motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure - High and two channels of Reactor Vessel Water Level - Low Low, Level 2 in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each Function. Thus, either two Reactor Vessel Water Level - Low Low, Level 2 or two Reactor Pressure - High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective motor breakers).

There are two motor breakers provided for each of the two recirculation pumps for a total of four breakers. The output of each trip system is provided to one of the two breakers for each recirculation pump.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The individual Functions are required to be OPERABLE in MODE 1 to protect against catastrophic/multiple failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure - High and Reactor Vessel Water Level - Low Low, Level 2 Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. Reactor Vessel Water Level - Low Low, Level 2 (LS-3-58A1, LS-3-58B1, LS-3-58C1, and LS-3-58D1)

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT System is initiated at Level 2 to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level - Low Low, Level 2 (LS-3-58A1, LS-3-58B1, LS-3-58C1, and LS-3-58D1) (continued)

Four channels of Reactor Vessel Water Level - Low Low, Level 2 with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation.

b. Reactor Steam Dome Pressure - High (PIS-3-204A, PIS-3-204B, PIS-3-204C, and PIS-3-204D)

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure - High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code limits.

The Reactor Steam Dome Pressure - High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor Steam Dome Pressure - High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure - High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code limits.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), and the Automatic Depressurization System (ADS). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS - Operating."

Portions of the ECCS instrumentation also provide for the generation of the Common Accident Signal which initiate the DGs and EECW System. Refer to LCO 3.8.1, "AC Systems-Operating," for operability requirements of the Common Accident Signal Logic.

Core Spray System

The CS System may be initiated by automatic means. Each pump can be controlled manually by a control room remote switch. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low, Level 1 or both Drywell Pressure - High and Reactor Steam Dome Pressure - Low. Reactor water level and drywell pressure are each monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of these trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function. The Reactor Steam Dome Pressure - Low variable is monitored by two transmitters for each trip system. The outputs from these transmitters are connected to relays arranged in a one-out-of-two logic.

(continued)

BASES

BACKGROUND

Core Spray System (continued)

The high drywell pressure and low reactor water level initiation signals are sealed in signals and must be manually reset. Upon receipt of an initiation signal, if normal AC power is available, the four core spray pumps start one at a time, in order, at 0, 7, 14, and 21 seconds. If normal AC power is not available, the four core spray pumps start seven seconds after standby power becomes available. (The LPCI pumps start as soon as standby power is available.)

The CS test line isolation valve is closed on a CS initiation signal to allow full system flow assumed in the accident analyses.

The CS pump discharge flow is monitored by a flow switch. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

The CS System logic also receives signals from transmitters which monitor the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. Reactor pressure is monitored by four redundant transmitters, which are, in turn, connected to four trip units (two per trip system). The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two logic for each trip system.

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low, Level 1 or both Drywell Pressure - High and Reactor Steam Dome Pressure - Low. Each of these diverse variables is monitored by four redundant transmitters, which, in turn, are connected to four trip units. The outputs of the trip units are connected to

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

Low reactor water level in the shroud is detected by two additional instruments which inhibit the manual initiation of other modes of RHR (e.g., suppression pool cooling) when LPCI is required. Manual overrides for the inhibit logic are provided.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low, Level 2 or Drywell Pressure - High. Each of these variables is monitored by four redundant transmitters, which are, in turn, connected to multiple trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

The HPCI pump discharge flow is monitored by a flow switch. Upon automatic initiation, when the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow, however, the flow rates assumed in the accident analysis can be achieved with the minimum flow valve open.

The HPCI test line isolation valve is closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis.

The HPCI System also monitors the water levels in the HPCI pump supply header from the condensate storage tank (CST) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in the HPCI pump supply header from the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

detect low water level in the HPCI pump supply header from the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close.

The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level - High, Level 8 trip, at which time the HPCI turbine trips, which causes the turbine's stop valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level - Low Low, Level 2 signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High or ADS High Drywell Pressure Bypass Timer; confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory); and CS or LPCI Pump Discharge Pressure - High are all present and the ADS Initiation Timer has timed out. There are two transmitters each for Reactor Vessel Water Level - Low Low Low, Level 1 and Drywell Pressure - High, and one transmitter for confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory) in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that

(continued)

BASES

BACKGROUND

Automatic Depressurization System (continued)

the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the ADS Initiation Timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

The ADS also monitors the discharge pressures of the four LPCI pumps and the four CS pumps. Each ADS trip system includes two discharge pressure permissive switches from all four LPCI pumps and one discharge pressure permissive switch from all four CS pumps. The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. CS pumps (A or B and either C or D) or any one of the four LPCI pumps is sufficient to permit automatic depressurization.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from each of the following variables: Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High; High Drywell Pressure Bypass Timer; and Pump Discharge Pressure - High. One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory). Either the Drywell Pressure - High or the Drywell Pressure Bypass Timer contacts and all remaining contacts in both logic strings must close and the ADS initiation timer must time out to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. Once the Drywell Pressure - High signal, the ADS High Drywell Pressure Bypass Timer, or the ADS initiation signal is present, it is individually sealed in until manually reset.

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS function, a combination of Functions is required to provide primary and secondary initiation signals.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D)

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level - Low Low Low, Level 1 is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating," for operability requirements of the Common Accident Signal Logic). The Reactor Vessel Water Level - Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Vessel Water Level - Low Low Low, Level 1 Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS - Shutdown," for Applicability Bases for the low pressure ECCS subsystems.

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating" for operability requirements of the Common Accident Signal Logic). The Drywell Pressure - High Function, along with the Reactor Steam Dome Pressure - Low Function, are directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

(continued)

BASES (continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are initiated by Functions 1.a, 1.b, 2.a, and 2.b (e.g., low pressure ECCS). The Required Action B.2 system would be HPCI. For Required Action B.1, redundant automatic initiation capability is lost if (a) two or more Function 1.a channels are inoperable and untripped such that both trip systems lose initiation capability, (b) two or more Function 2.a channels are inoperable and untripped such that both trip systems lose initiation capability, (c) two or more Function 1.b channels are inoperable and untripped such that both trip systems lose initiation capability, or (d) two or more Function 2.b channels are inoperable and untripped such that both trip systems lose initiation capability. For low pressure ECCS, since each inoperable

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

either (a) four Function 1.c channels are inoperable (i.e., both channels in both trip systems are inoperable), (b) two or more Function 2.c channels are inoperable such that both trip systems lose initiation capability, (c) two or more Function 2.d channels are inoperable such that both trip systems lose initiation capability, (d) one or more Function 1.e channels are inoperable in both trip systems (i.e., at least one CS pump in both subsystems is affected), or (e) multiple Function 2.f channels are inoperable such that the trip systems cannot start both LPCI pumps in at least one subsystem. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action C.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for both low pressure ECCS subsystems are inoperable (e.g., both CS subsystems), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For Functions 1.c, 2.c, 1.e, 2.d, and 2.f, the affected portions are the associated low pressure ECCS pumps. As noted (Note 1), Required Action C.1 is only applicable in MODES 1, 2, and 3.

In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of automatic initiation capability for 24 hours (as allowed by Required Action C.2) is allowed during MODES 4 and 5.

Note 2 states that Required Action C.1 is only applicable for Functions 1.c, 2.c, 1.e, 2.d, and 2.f. Required Action C.1 is also not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable). The loss of one Function 3.c channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference 4 and considered acceptable for the 24 hours allowed by Required Action C.2.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that redundant features in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events.

D.1

Required Action D.1 is intended to ensure that appropriate actions are taken if an inoperable, untripped channel within the same Function results in a complete loss of automatic component initiation capability for the HPCI System. Since Table 3.3.5.1-1 only requires one channel to be OPERABLE, automatic component initiation capability is lost if the one required Function 3.d channel or the one required Function 3.e channel is inoperable and untripped. In this situation (loss of automatic suction swap), the HPCI system must be declared inoperable within 1 hour. As noted, Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

(continued)

BASES

ACTIONS

D.1 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the Core Spray Pump Discharge Flow - Low Bypass Function results in redundant automatic initiation capability being lost for the feature(s). Automatic initiation capability of the Core Spray Pump Discharge Flow - Low (Bypass) Function in both CS subsystems is lost if two Function 1.d channels are inoperable. In this situation (loss of capability for both subsystems), the 7 day allowance of Required Action E.2 is not appropriate and the subsystem associated with each inoperable channel must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected CS pump to be declared inoperable. However, since channels for both CS subsystems are inoperable, and the completion times started concurrently for both channels this results in all four CS pumps being concurrently declared inoperable. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to Function 1.d. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 4 and considered acceptable for the 7 days allowed by Required Action E.2.

(continued)

BASES

ACTIONS (continued)

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 6 hours.

(continued)

BASES

ACTIONS

E.1 (continued)

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

F.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channels.

For the RWCU Area Temperature - High Functions, the affected penetration flow path(s) may be considered isolated by isolating only that portion of the system in the associated room monitored by the inoperable channel. That is, if the RWCU pump room A area channel is inoperable, the pump room A area can be isolated while allowing continued RWCU operation utilizing the B RWCU pump.

Alternately, if it is not desired to isolate the affected penetration flow path(s) (e.g., as in the case where isolating the penetration flow path(s) could result in a reactor scram), Condition G must be entered and its Required Actions taken.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for plant operations personnel to isolate the affected penetration flow path(s).

G.1 and G.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, or any Required Action of Condition F is not met and the associated Completion Time has expired, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1 and H.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the SLC System is declared inoperable or the RWCU System is isolated. Since this Function is required to ensure that the SLC System performs its intended function, sufficient remedial measures are provided by declaring the SLC System inoperable or isolating the RWCU System.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for personnel to isolate the RWCU System.

I.1 and I.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path(s) should be closed. However, if the shutdown cooling function is needed to provide core cooling, these Required Actions allow the penetration flow path(s) to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path(s) can be isolated). Actions must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each Primary Containment Isolation instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 5 and 6) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the PCIVs will isolate the penetration flow path(s) when necessary.

SR 3.3.6.1.1

Performance of the CHANNEL CHECK once every 24 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 instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff 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 instrument has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.1 (continued)

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 the channels required by the LCO.

SR 3.3.6.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analysis described in References 5 and 6.

SR 3.3.6.1.3, SR 3.3.6.1.4 and SR 3.3.6.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies of SR 3.3.6.1.3, SR 3.3.6.1.4, and SR 3.3.6.1.5 are based on the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The LOGIC SYSTEM FUNCTIONAL TEST shall include a calibration of time delay relays and timers necessary for proper functioning of the logic. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown these components usually pass the Surveillance when performed at the Frequency provided.

REFERENCES

1. FSAR, Section 6.5.

2. FSAR, Chapter 14.

3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

4. FSAR, Section 4.9.3.

5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

(continued)

BASES

REFERENCES (continued)

7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

8. FSAR, Section 5.2.

9. NRC letter from Richard J. Clark to Hugh G. Parris dated August 9, 1984, Safety Evaluation for Amendment Nos. 107, 101, and 74 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant Units 1, 2, and 3 respectively.

BASES

ACTIONS

E.1 and E.2 (continued)

placed in operation (Required Action E.1) while the other CREV subsystem can be declared inoperable (Required Action E.2).

The 1 hour Completion Time is intended to allow the operator time to place the CREV subsystem(s) in operation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, for placing the associated CREV subsystem(s) in operation, or for entering the applicable Conditions and Required Actions for the inoperable CREV subsystem(s).

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each CREV System instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CREV System initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 3 and 4) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the CREV System will initiate when necessary.

The Surveillances are modified by a third Note (Note 3) to indicate that for Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of a CHANNEL CALIBRATION or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the tripped condition. Upon completion of the Surveillance or maintenance, or expiration of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

24 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

SR 3.3.7.1.1

Performance of the CHANNEL CHECK once every 24 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 instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, 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 instrument has drifted outside its limit.

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

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The LOP instrumentation monitors the 4.16 kV shutdown boards. Offsite power is the preferred source of power for the 4.16 kV shutdown boards. If the monitors determine that insufficient power is available, the boards are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources.

Each 4.16 kV shutdown board has its own independent LOP instrumentation and associated trip logic. The voltage for each board is monitored at two levels, which can be considered as two different undervoltage Functions: Loss of Voltage and 4.16 kV Shutdown Board Undervoltage Degraded Voltage. Each Function causes various board transfers and disconnects.

The Degraded Voltage Function is monitored by three undervoltage relay channels for each shutdown board, whose outputs are arranged in a two-out-of-three logic configuration (Ref. 1). The channels compare measured input signals with pre-established setpoints. When the setpoint is exceeded for two-of-three degraded voltage channels, the logic energizes timers which provides a LOP trip signal to the shutdown board logic.

The Loss of Voltage Function is monitored by two undervoltage relay pairs for each shutdown board, where outputs are arranged in a two-out-of-two logic configuration (Ref. 1). The channels include four electro-mechanical relays, two of which must deenergize to start the associated diesel generator and another two which must deenergize to initiate load shed of the associated 4.16 kV shutdown board.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for unit channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis. The channel devices for each shutdown board are listed in Table B 3.3.8.1-1.

1. 4.16 kV Shutdown Board Undervoltage (Loss of Voltage)

Loss of voltage on a 4.16 kV shutdown board indicates that offsite power may be completely lost to the respective shutdown board and is unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the board is transferred from offsite power to DG power upon total loss of shutdown board voltage for 1.5 seconds. The transfer will not occur if the voltage recovers to the specified Allowable Value for Reset Voltage within 1.5 seconds. This ensures that adequate power will be available to the required equipment.

The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that power is available to the required equipment.

Two channels of 4.16 kV Shutdown Board Undervoltage (Loss of Voltage) Function per associated shutdown board are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1, "AC Sources - Operating," and 3.8.2, "AC Sources - Shutdown," for Applicability Bases for the DGs.

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BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV shutdown board indicates that, while offsite power may not be completely lost to the respective shutdown board, available power maybe insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the board is transferred from offsite power to onsite DG power when the voltage on the board drops below the Degraded Voltage Function Allowable Values (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The Board Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

Three channels of 4.16 kV Shutdown Board Undervoltage (Degraded Voltage) Function per associated board are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS

A Note has been provided to modify the ACTIONS related to LOP instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable LOP instrumentation channels provide appropriate compensatory measures for separate inoperable channels.

(continued)

BASES

ACTIONS (continued)

As such, a Note has been provided that allows separate Condition entry for each inoperable LOP instrumentation channel.

A.1 and A.2

With one of the degraded voltage relay channels inoperable, Required Action A.2 provides a 15 day allowable out of service time to restore the relay channel to OPERABLE status provided the other two degraded voltage relay channels and associated timers and the loss of voltage relay channels on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of the other degraded voltage relay channels and associated timers and loss of voltage relay channels is therefore required (Required Action A.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition C or D, as applicable, must be entered immediately. The 15 day allowable out of service time is justified based on the two-out-of-three permissive logic scheme provided for these relays. If the inoperable relay channel cannot be restored to OPERABLE status within the allowable out of service time, the degraded voltage relay channel must be placed in the tripped condition per Required Action A.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

B.1 and B.2

With one or more loss of voltage relay channels inoperable, the Function is not capable of performing the intended function. Required Action B.2 provides a 10 day allowable out of service time provided two or more degraded voltage

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

relay channels and associated timers on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of two or more degraded voltage relay channels and associated timers is therefore required (Required Action B.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition D must be entered immediately. The 10 day allowable out of service time is justified since the degraded voltage relay channels on the same shutdown board are independent of the loss of voltage relay channels and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channels cannot be restored to OPERABLE status within the allowable out of service time, the channel(s) must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel(s) in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel(s) in trip (e.g., as in the case where placing the channel(s) in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

C.1 and C.2

With two or more degraded voltage relay channels or one or more associated timers inoperable, the Function is not capable of performing the intended function. Required Action C.2 provides a 10 day allowable out of service time provided the loss of voltage relay channels on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of the loss of voltage relay channels is therefore required (Required Action C.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

equipment cannot be verified however, Condition D must be entered immediately.

The 10 day allowable out of service time is justified since the loss of voltage relay channels on the same shutdown board are independent of the degraded voltage relay channel(s) and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channel(s) cannot be restored to OPERABLE status within the allowable out of service time, the channel(s) must be placed in the tripped condition per Required Action C.2. Placing the inoperable channel(s) in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel(s) in trip (e.g., as in the case where placing the channel(s) in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

D.1 and D.2

With two or more degraded voltage relay channels or one or more associated timers and the loss of voltage relay channel(s) inoperable on the same shutdown board, the associated diesel generator will not automatically start upon degraded voltage or complete loss of voltage on that shutdown board. In this situation, Required Action D.2 provides a 5 day allowable out of service time provided the other shutdown boards and undervoltage relay channels are OPERABLE. Immediate verification of the OPERABILITY of the other shutdown boards and undervoltage relay channels is therefore required (Required Action D.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition E must be entered immediately. The 5 day allowable out of service time is justified based on the remaining redundancy of the 4.16 kV Shutdown Boards. The 4.16 kV Shutdown Boards have a

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

similar allowable out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

E.1

If any Required Action and associated Completion Time are not met, the associated Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each LOP instrumentation Function are located in the SRs column of Table 3.3.8.1-1.

SR 3.3.8.1.1 and SR 3.3.8.1.2

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1.1 and SR 3.3.8.1.2 (continued)

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency is based upon the calibration interval assumed in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.1.3

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety functions.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Figure 8.4-4.
2. FSAR, Section 6.5.
3. FSAR, Section 8.5.4.
4. FSAR, Chapter 14.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 2 ITS BASES (Revised pages marked *R2)

Replaced B 3.3-1 with B 3.3-1 Revision 2
Replaced B 3.3-8 Revision 1 with B 3.3-8 Revision 2
Replaced B 3.3-25 Revision 1 with B 3.3-25 Revision 2
Replaced B 3.3-31 Revision 1 with B 3.3-31 Revision 2
Replaced B 3.3-32 Revision 1 with B 3.3-32 Revision 2
Replaced B 3.3-46 Revision 1 with B 3.3-46 Revision 2
Replaced B 3.3-80 Revision 1 with B 3.3-80 Revision 2
Replaced B 3.3-92 Revision 1 with B 3.3-92 Revision 2
Replaced B 3.3-94 Revision 1 with B 3.3-94 Revision 2
Replaced B 3.3-95 Revision 1 with B 3.3-95 Revision 2
Replaced B 3.3-101 Revision 1 with B 3.3-101 Revision 2
Replaced B 3.3-102 Revision 1 with B 3.3-102 Revision 2
Replaced B 3.3-104 Revision 1 with B 3.3-104 Revision 2
Replaced B 3.3-105 Revision 1 with B 3.3-105 Revision 2
Replaced B 3.3-106 Revision 1 with B 3.3-106 Revision 2
Replaced B 3.3-108 Revision 1 with B 3.3-108 Revision 2
Replaced B 3.3-109 Revision 1 with B 3.3-109 Revision 2
Replaced B 3.3-125 Revision 1 with B 3.3-125 Revision 2
Replaced B 3.3-128 Revision 1 with B 3.3-128 Revision 2
Replaced B 3.3-129 Revision 1 with B 3.3-129 Revision 2
Replaced B 3.3-130 Revision 1 with B 3.3-130 Revision 2
Replaced B 3.3-168 Revision 1 with B 3.3-168 Revision 2
Replaced B 3.3-169 Revision 1 with B 3.3-169 Revision 2
Replaced B 3.3-170 Revision 1 with B 3.3-170 Revision 2
Replaced B 3.3-171 Revision 1 with B 3.3-171 Revision 2
Replaced B 3.3-172 Revision 1 with B 3.3-172 Revision 2
Replaced B 3.3-173 Revision 1 with B 3.3-173 Revision 2
Replaced B 3.3-174 Revision 1 with B 3.3-174 Revision 2
Replaced B 3.3-196 Revision 1 with B 3.3-196 Revision 2
Replaced B 3.3-197 Revision 1 with B 3.3-197 Revision 2
Replaced B 3.3-200 Revision 1 with B 3.3-200 Revision 2
Replaced B 3.3-202 Revision 1 with B 3.3-202 Revision 2
Replaced B 3.3-203 Revision 1 with B 3.3-203 Revision 2
Replaced B 3.3-204 Revision 1 with B 3.3-204 Revision 2
Replaced B 3.3-205 Revision 1 with B 3.3-205 Revision 2
Replaced B 3.3-206 Revision 1 with B 3.3-206 Revision 2
Replaced B 3.3-207 Revision 1 with B 3.3-207 Revision 2
Replaced B 3.3-208 Revision 1 with B 3.3-208 Revision 2

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure 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 LCOs on other reactor system parameters and equipment performance. The LSSS are defined in this Specification as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).

The RPS, as described in the FSAR, Section 7.2 (Ref. 1), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure trip oil pressure (indicated by TCV low hydraulic pressure), turbine stop valve (TSV) position, drywell pressure, scram pilot air header pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position, manual, and RPS channel test switch scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown, manual, and RPS channel test switch scram signals). Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay deenergizes, which then outputs an RPS trip signal to the trip logic.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux - High, Setdown (continued)

14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve - Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).

For Function 8 (Turbine Stop Valve - Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip).

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/T average core exposure Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the scram pilot air header low pressure trip function is based on the functional reliability previously demonstrated by this function, the need for minimizing the radiation exposure associated with the functional testing of this function, and the increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by the more frequent functional testing.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For MSIV-Closure, SDV Water Level - High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 includes physical inspection and actuation of the switches.

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from 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 performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint are appropriately compared to a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 29% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.75, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.44, no RWE event will result in exceeding the MCPR

(continued)

Table B 3.3.3.2-1 (Page 1 of 3)
Backup Control System Instrumentation and Controls

FUNCTION	NUMBER REQUIRED
Instrument Parameter	
1. Reactor Water Level Indication	1
2. Reactor Pressure Indication	1
3. Suppression Pool Temperature Indication	1
4. Suppression Pool Level Indication	1
5. Drywell Pressure Indication	1
6. RHR Flow Indication	1
7. RCIC Flow Indication	1, note a
8. RCIC Turbine Speed Indication	1
9. Drywell Temperature Indication	1
10. RHRSW Header Pressure	1, note p
Transfer/Control Parameter	
11. Main Steam Relief Valve (MSRV) Transfer & Control	3, note b
12. Main Steam Isolation Valve (MSIV) Transfer & Control (Closure)	4, note c
13. Main Steam Drain Line Isolation Valve	1, note d
14. RHRSW Pumps	note e
15. RHRSW Discharge Valves for RHR Loop I Heat Exchangers	2, note f
note a: RCIC flow indication may be obtained from the Flow Indicating Controller	
note b: 1 required for each of 3 MSRVs.	
note c: 1 MSIV required per penetration, may be either inboard valve or outboard valve.	
note d: 1 Main Steam Drain Line isolation valve required, may be either inboard valve or outboard valve.	
note e: There are 12 RHRSW pumps. All are equipped with emergency transfer switches. 2 of the 12 must be available for EECW service (supports all units) and an additional 1 must be available for RHRSW service.	
note f: 1 Discharge Valve per RHR Loop I Heat Exchanger for a total of 2.	
note o: Note not used.	
note p: The RHRSW Pressure indicator for the Header of the RHRSW Pump that supports RHR service is required.	

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level - Low Low, Level 2 or Reactor Steam Dome Pressure - High setpoint is reached, the recirculation pump motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure - High and two channels of Reactor Vessel Water Level - Low Low, Level 2 in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each Function. Thus, either two Reactor Vessel Water Level - Low Low, Level 2 or two Reactor Pressure - High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective motor breakers).

There are two motor breakers provided for each of the two recirculation pumps for a total of four breakers. The output of each trip system is provided to one of the two breakers for each recirculation pump.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The individual Functions are required to be OPERABLE in MODE 1 to protect against catastrophic/multiple failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure - High and Reactor Vessel Water Level - Low Low, Level 2 Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. Reactor Vessel Water Level - Low Low, Level 2 (LS-3-58A1, LS-3-58B1, LS-3-58C1, and LS-3-58D1)

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT System is initiated at Level 2 to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level - Low Low, Level 2 (LS-3-58A1, LS-3-58B1, LS-3-58C1, and LS-3-58D1) (continued)

Four channels of Reactor Vessel Water Level - Low Low, Level 2 with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation.

b. Reactor Steam Dome Pressure - High (PIS-3-204A, PIS-3-204B, PIS-3-204C, and PIS-3-204D)

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure - High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code limits.

The Reactor Steam Dome Pressure - High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor Steam Dome Pressure - High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure - High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code limits.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), and the Automatic Depressurization System (ADS). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS - Operating."

Portions of the ECCS instrumentation also provide for the generation of the Common Accident Signal which initiate the DGs and EECW System. Refer to LCO 3.8.1, "AC Systems-Operating," for operability requirements of the Common Accident Signal Logic.

Core Spray System

The CS System may be initiated by automatic means. Each pump can be controlled manually by a control room remote switch. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low, Level 1 or both Drywell Pressure - High and Reactor Steam Dome Pressure - Low. Reactor water level and drywell pressure are each monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of these trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function. The Reactor Steam Dome Pressure - Low variable is monitored by two transmitters for each trip system. The outputs from these transmitters are connected to relays arranged in a one-out-of-two logic.

(continued)

BASES

BACKGROUND

Core Spray System (continued)

The high drywell pressure and low reactor water level initiation signals are sealed in signals and must be manually reset. Upon receipt of an initiation signal, if normal AC power is available, the four core spray pumps start one at a time, in order, at 0, 7, 14, and 21 seconds. If normal AC power is not available, the four core spray pumps start seven seconds after standby power becomes available. (The LPCI pumps start as soon as standby power is available.)

The CS test line isolation valve is closed on a CS initiation signal to allow full system flow assumed in the accident analyses.

The CS pump discharge flow is monitored by a flow switch. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

The CS System logic also receives signals from transmitters which monitor the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. Reactor pressure is monitored by four redundant transmitters, which are, in turn, connected to four trip units (two per trip system). The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two logic for each CS trip system.

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low, Level 1 or both Drywell Pressure - High and Reactor Steam Dome Pressure - Low. Each of these diverse variables is monitored by four redundant transmitters, which, in turn, are connected to four trip

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

Low reactor water level in the shroud is detected by two additional instruments which inhibit the manual initiation of other modes of RHR (e.g., suppression pool cooling) when LPCI is required. Manual overrides for the inhibit logic are provided.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low, Level 2 or Drywell Pressure - High. Each of these variables is monitored by four redundant transmitters, which are, in turn, connected to multiple trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

The HPCI pump discharge flow is monitored by a flow switch. Upon automatic initiation, when the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow, however, the flow rates assumed in the accident analysis can be achieved with the minimum flow valve open.

The HPCI test line isolation valve is closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis.

The HPCI System also monitors the water levels in the HPCI pump supply header from the condensate storage tank (CST) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in the HPCI pump supply header from the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

detect low water level in the HPCI pump supply header from the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close.

The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level - High, Level 8 trip, at which time the HPCI turbine trips, which causes the turbine's stop valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level - Low Low, Level 2 signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High or ADS High Drywell Pressure Bypass Timer; confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory); and CS or LPCI Pump Discharge Pressure - High are all present and the ADS Initiation Timer has timed out. There are two transmitters each for Reactor Vessel Water Level - Low Low Low, Level 1 and Drywell Pressure - High, and one transmitter for confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory) in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that

(continued)

BASES

BACKGROUND

Automatic Depressurization System (continued)

the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the ADS Initiation Timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

The ADS also monitors the discharge pressures of the four LPCI pumps and the four CS pumps. Each ADS trip system includes two discharge pressure permissive switches from all four LPCI pumps and one discharge pressure permissive switch from all four CS pumps. The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. CS pumps (A or B and either C or D) or any one of the four LPCI pumps is sufficient to permit automatic depressurization.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from each of the following variables: Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High; High Drywell Pressure Bypass Timer; and Pump Discharge Pressure - High. One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory). Either the Drywell Pressure - High or the Drywell Pressure Bypass Timer contacts and all remaining contacts in both logic strings must close and the ADS initiation timer must time out to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. Once the Drywell Pressure - High signal, the ADS High Drywell Pressure Bypass Timer, or the ADS initiation signal is present, it is individually sealed in until manually reset.

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS function, a combination of Functions is required to provide primary and secondary initiation signals.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D)

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level - Low Low Low, Level 1 is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating," for operability requirements of the Common Accident Signal Logic). The Reactor Vessel Water Level - Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Vessel Water Level - Low Low Low, Level 1 Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS - Shutdown," for Applicability Bases for the low pressure ECCS subsystems.

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating" for operability requirements of the Common Accident Signal Logic). The Drywell Pressure - High Function, along with the Reactor Steam Dome Pressure - Low Function, are directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

(continued)

BASES (continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are initiated by Functions 1.a, 1.b, 2.a, and 2.b (e.g., low pressure ECCS). The Required Action B.2 system would be HPCI. For Required Action B.1, redundant automatic initiation capability is lost if (a) two or more Function 1.a channels are inoperable and untripped such that both trip systems lose initiation capability, (b) two or more Function 2.a channels are inoperable and untripped such that both trip systems lose initiation capability, (c) two or more Function 1.b channels are inoperable and untripped such that both trip systems lose initiation capability, or (d) two or more Function 2.b channels are inoperable and untripped such that both trip systems lose initiation capability. For low pressure ECCS, since each inoperable

(continued)

BASES

ACTIONS

C.1, and C.2 (continued)

either (a) four Function 1.c channels are inoperable (i.e., both channels in both trip systems are inoperable), (b) two or more Function 2.c channels are inoperable such that both trip systems lose initiation capability, (c) two or more Function 2.d channels are inoperable such that both trip systems lose initiation capability, (d) one or more Function 1.e channels are inoperable in both trip systems (i.e., at least one CS pump in both subsystems is affected), or (e) multiple Function 2.f channels are inoperable such that the trip systems cannot start both LPCI pumps in at least one subsystem. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action C.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for both low pressure ECCS subsystems are inoperable (e.g., both CS subsystems), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For Functions 1.c, 2.c, 1.e, 2.d, and 2.f, the affected portions are the associated low pressure ECCS pumps. As noted (Note 1), Required Action C.1 is only applicable in MODES 1, 2, and 3.

In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of automatic initiation capability for 24 hours (as allowed by Required Action C.2) is allowed during MODES 4 and 5.

Note 2 states that Required Action C.1 is only applicable for Functions 1.c, 2.c, 1.e, 2.d, and 2.f. Required Action C.1 is also not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable). The loss of one Function 3.c channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference 4 and considered acceptable for the 24 hours allowed by Required Action C.2.

(continued)

BASES

ACTIONS

C.1, and C.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that redundant features in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events.

D.1

Required Action D.1 is intended to ensure that appropriate actions are taken if an inoperable, untripped channel within the same Function results in a complete loss of automatic component initiation capability for the HPCI System. Since Table 3.3.5.1-1 only requires one channel to be OPERABLE, automatic component initiation capability is lost if the one required Function 3.d channel or the one required Function 3.e channel is inoperable and untripped. In this situation (loss of automatic suction swap), the HPCI system must be declared inoperable within 1 hour. As noted, Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

(continued)

BASES

ACTIONS

D.1 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the Core Spray Pump Discharge Flow - Low Bypass Function results in redundant automatic initiation capability being lost for the feature(s). Automatic initiation capability of the Core Spray Pump Discharge Flow - Low (Bypass) Function in both CS subsystems is lost if two Function 1.d channels are inoperable. In this situation (loss of capability for both subsystems), the 7 day allowance of Required Action E.2 is not appropriate and the subsystem associated with each inoperable channel must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected CS pump to be declared inoperable. However, since channels for both CS subsystems are inoperable, and the completion times started concurrently for both channels this results in all four CS pumps being concurrently declared inoperable. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to Function 1.d. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 4 and considered acceptable for the 7 days allowed by Required Action E.2.

(continued)

BASES

ACTIONS (continued)

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 6 hours.

(continued)

BASES

ACTIONS

E.1 (continued)

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

F.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channels.

For the RWCU Area Temperature - High Functions, the affected penetration flow path(s) may be considered isolated by isolating only that portion of the system in the associated room monitored by the inoperable channel. That is, if the RWCU pump room A area channel is inoperable, the pump room A area can be isolated while allowing continued RWCU operation utilizing the B RWCU pump.

Alternately, if it is not desired to isolate the affected penetration flow path(s) (e.g., as in the case where isolating the penetration flow path(s) could result in a reactor scram), Condition G must be entered and its Required Actions taken.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for plant operations personnel to isolate the affected penetration flow path(s).

G.1 and G.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, or any Required Action of Condition F is not met and the associated Completion Time has expired, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1 and H.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the SLC System is declared inoperable or the RWCU System is isolated. Since this Function is required to ensure that the SLC System performs its intended function, sufficient remedial measures are provided by declaring the SLC System inoperable or isolating the RWCU System.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for personnel to isolate the RWCU System.

I.1 and I.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path(s) should be closed. However, if the shutdown cooling function is needed to provide core cooling, these Required Actions allow the penetration flow path(s) to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path(s) can be isolated). Actions must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each Primary Containment Isolation instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 5 and 6) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the PCIVs will isolate the penetration flow path(s) when necessary.

SR 3.3.6.1.1

Performance of the CHANNEL CHECK once every 24 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 instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff 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 instrument has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.1 (continued)

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 the channels required by the LCO.

SR 3.3.6.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analysis described in References 5 and 6.

SR 3.3.6.1.3, SR 3.3.6.1.4 and SR 3.3.6.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies of SR 3.3.6.1.3, SR 3.3.6.1.4, and SR 3.3.6.1.5 are based on the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The LOGIC SYSTEM FUNCTIONAL TEST shall include a calibration of time delay relays and timers necessary for proper functioning of the logic. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown these components usually pass the Surveillance when performed at the Frequency provided.

REFERENCES

1. FSAR, Section 6.5.

2. FSAR, Chapter 14.

3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

4. FSAR, Section 4.9.3.

5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

(continued)

BASES

REFERENCES (continued)

7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

8. FSAR, Section 5.2.

9. NRC letter from Richard J. Clark to Hugh G. Parris dated August 9, 1984, Safety Evaluation for Amendment Nos. 107, 101, and 74 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant Units 1, 2, and 3 respectively.

BASES

ACTIONS

E.1 and E.2 (continued)

placed in operation (Required Action E.1) while the other CREV subsystem can be declared inoperable (Required Action E.2).

The 1 hour Completion Time is intended to allow the operator time to place the CREV subsystem(s) in operation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, for placing the associated CREV subsystem(s) in operation, or for entering the applicable Conditions and Required Actions for the inoperable CREV subsystem(s).

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each CREV System instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CREV System initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 3 and 4) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the CREV System will initiate when necessary.

The Surveillances are modified by a third Note (Note 3) to indicate that for Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of a CHANNEL CALIBRATION or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the tripped condition. Upon completion of the Surveillance or maintenance, or expiration of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

24 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

SR 3.3.7.1.1

Performance of the CHANNEL CHECK once every 24 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 instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, 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 instrument has drifted outside its limit.

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

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The LOP instrumentation monitors the 4.16 kV shutdown boards. Offsite power is the preferred source of power for the 4.16 kV shutdown boards. If the monitors determine that insufficient power is available, the boards are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources.

Each 4.16 kV shutdown board has its own independent LOP instrumentation and associated trip logic. The voltage for each board is monitored at two levels, which can be considered as two different undervoltage Functions: Loss of Voltage and 4.16 kV Shutdown Board Undervoltage Degraded Voltage. Each Function causes various board transfers and disconnects.

The Degraded Voltage Function is monitored by three undervoltage relay channels for each shutdown board, whose outputs are arranged in a two-out-of-three logic configuration (Ref. 1). The channels compare measured input signals with pre-established setpoints. When the setpoint is exceeded for two-of-three degraded voltage channels, the logic energizes timers which provides a LOP trip signal to the shutdown board logic.

The Loss of Voltage Function is monitored by two undervoltage relay pairs for each shutdown board, where outputs are arranged in a two-out-of-two logic configuration (Ref. 1). The channels include four electro-mechanical relays, two of which must deenergize to start the associated diesel generator and another two which must deenergize to initiate load shed of the associated 4.16 kV shutdown board.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for unit channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis. The channel devices for each shutdown board are listed in Table B 3.3.8.1-1.

1. 4.16 kV Shutdown Board Undervoltage (Loss of Voltage)

Loss of voltage on a 4.16 kV shutdown board indicates that offsite power may be completely lost to the respective shutdown board and is unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the board is transferred from offsite power to DG power upon total loss of shutdown board voltage for 1.5 seconds. The transfer will not occur if the voltage recovers to the specified Allowable Value for Reset Voltage within 1.5 seconds. This ensures that adequate power will be available to the required equipment.

The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that power is available to the required equipment.

Two channels of 4.16 kV Shutdown Board Undervoltage (Loss of Voltage) Function per associated shutdown board are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1, "AC Sources - Operating," and 3.8.2, "AC Sources - Shutdown," for Applicability Bases for the DGs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES LCO, and APPLICABILITY (continued)

2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV shutdown board indicates that, while offsite power may not be completely lost to the respective shutdown board, available power maybe insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the board is transferred from offsite power to onsite DG power when the voltage on the board drops below the Degraded Voltage Function Allowable Values (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The Board Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

Three channels of 4.16 kV Shutdown Board Undervoltage (Degraded Voltage) Function per associated board are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS

A Note has been provided to modify the ACTIONS related to LOP instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable LOP instrumentation channels provide appropriate compensatory measures for separate inoperable channels.

(continued)

BASES

ACTIONS (continued)

As such, a Note has been provided that allows separate Condition entry for each inoperable LOP instrumentation channel.

A.1 and A.2

With one of the degraded voltage relay channels inoperable, Required Action A.2 provides a 15 day allowable out of service time to restore the relay channel to OPERABLE status provided the other two degraded voltage relay channels and associated timers and the loss of voltage relay channels on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of the other degraded voltage relay channels and associated timers and loss of voltage relay channels is therefore required (Required Action A.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition C or D, as applicable, must be entered immediately. The 15 day allowable out of service time is justified based on the two-out-of-three permissive logic scheme provided for these relays. If the inoperable relay channel cannot be restored to OPERABLE status within the allowable out of service time, the degraded voltage relay channel must be placed in the tripped condition per Required Action A.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

B.1 and B.2

With one or more loss of voltage relay channels inoperable, the Function is not capable of performing the intended function. Required Action B.2 provides a 10 day allowable out of service time provided two or more degraded voltage

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

relay channels and associated timers on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of two or more degraded voltage relay channels and associated timers is therefore required (Required Action B.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition D must be entered immediately. The 10 day allowable out of service time is justified since the degraded voltage relay channels on the same shutdown board are independent of the loss of voltage relay channels and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channels cannot be restored to OPERABLE status within the allowable out of service time, the channel(s) must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel(s) in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel(s) in trip (e.g., as in the case where placing the channel(s) in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

C.1 and C.2

With two or more degraded voltage relay channels or one or more associated timers inoperable, the Function is not capable of performing the intended function. Required Action C.2 provides a 10 day allowable out of service time provided the loss of voltage relay channels on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of the loss of voltage relay channels is therefore required (Required Action C.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

equipment cannot be verified however, Condition D must be entered immediately.

The 10 day allowable out of service time is justified since the loss of voltage relay channels on the same shutdown board are independent of the degraded voltage relay channel(s) and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channel(s) cannot be restored to OPERABLE status within the allowable out of service time, the channel(s) must be placed in the tripped condition per Required Action C.2. Placing the inoperable channel(s) in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel(s) in trip (e.g., as in the case where placing the channel(s) in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

D.1 and D.2

With two or more degraded voltage relay channels or one or more associated timers and the loss of voltage relay channel(s) inoperable on the same shutdown board, the associated diesel generator will not automatically start upon degraded voltage or complete loss of voltage on that shutdown board. In this situation, Required Action D.2 provides a 5 day allowable out of service time provided the other shutdown boards and undervoltage relay channels are OPERABLE. Immediate verification of the OPERABILITY of the other shutdown boards and undervoltage relay channels is therefore required (Required Action D.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition E must be entered immediately. The 5 day allowable out of service time is justified based on the remaining redundancy of the 4.16 kV Shutdown Boards. The 4.16 kV Shutdown Boards have a

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

similar allowable out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

E.1

If any Required Action and associated Completion Time are not met, the associated Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each LOP instrumentation Function are located in the SRs column of Table 3.3.8.1-1.

SR 3.3.8.1.1 and SR 3.3.8.1.2

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1.1 and SR 3.3.8.1.2 (continued)

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency is based upon the calibration interval assumed in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.1.3

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety functions.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Figure 8.4-4.
2. FSAR, Section 6.5.
3. FSAR, Section 8.5.4.
4. FSAR, Chapter 14.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 3 ITS BASES (Revised pages marked *R1)

Replaced B 3.3-1 with B 3.3-1 Revision 2
Replaced B 3.3-8 Revision 1 with B 3.3-8 Revision 2
Replaced B 3.3-25 Revision 1 with B 3.3-25 Revision 2
Replaced B 3.3-31 Revision 1 with B 3.3-31 Revision 2
Replaced B 3.3-32 Revision 1with B 3.3-32 Revision 2
Replaced B 3.3-46 Revision 1with B 3.3-46 Revision 2
Replaced B 3.3-79 Revision 1with B 3.3-79 Revision 2
Replaced B 3.3-80 Revision 1with B 3.3-80 Revision 2
Replaced B 3.3-92 Revision 1 with B 3.3-92 Revision 2
Replaced B 3.3-94 Revision 1 with B 3.3-94 Revision 2
Replaced B 3.3-95 Revision 1 with B 3.3-95 Revision 2
Replaced B 3.3-101 Revision 1 with B 3.3-101 Revision 2
Replaced B 3.3-102 Revision 1 with B 3.3-102 Revision 2
Replaced B 3.3-104 Revision 1 with B 3.3-104 Revision 2
Replaced B 3.3-105 Revision 1 with B 3.3-105 Revision 2
Replaced B 3.3-106 Revision 1 with B 3.3-106 Revision 2
Replaced B 3.3-108 Revision 1 with B 3.3-108 Revision 2
Replaced B 3.3-109 Revision 1 with B 3.3-109 Revision 2
Replaced B 3.3-125 Revision 1 with B 3.3-125 Revision 2
Replaced B 3.3-127 Revision 1 with B 3.3-127 Revision 2
Replaced B 3.3-128 Revision 1 with B 3.3-128 Revision 2
Replaced B 3.3-129 Revision 1 with B 3.3-129 Revision 2
Replaced B 3.3-130 Revision 1 with B 3.3-130 Revision 2
Replaced B 3.3-168 Revision 1 with B 3.3-168 Revision 2
Replaced B 3.3-169 Revision 1 with B 3.3-169 Revision 2
Replaced B 3.3-170 Revision 1 with B 3.3-170 Revision 2
Replaced B 3.3-171 Revision 1 with B 3.3-171 Revision 2
Replaced B 3.3-172 Revision 1 with B 3.3-172 Revision 2
Replaced B 3.3-173 Revision 1 with B 3.3-173 Revision 2
Replaced B 3.3-174 Revision 1 with B 3.3-174 Revision 2
Replaced B 3.3-196 Revision 1 with B 3.3-196 Revision 2
Replaced B 3.3-197 Revision 1 with B 3.3-197 Revision 2
Replaced B 3.3-200 Revision 1 with B 3.3-200 Revision 2
Replaced B 3.3-202 Revision 1 with B 3.3-202 Revision 2
Replaced B 3.3-203 Revision 1 with B 3.3-203 Revision 2
Replaced B 3.3-204 Revision 1 with B 3.3-204 Revision 2
Replaced B 3.3-205 Revision 1 with B 3.3-205 Revision 2
Replaced B 3.3-206 Revision 1 with B 3.3-206 Revision 2
Replaced B 3.3-207 Revision 1 with B 3.3-207 Revision 2
Replaced B 3.3-208 Revision 1 with B 3.3-208 Revision 2

B 3.3 INSTRUMENTATION

B 3.3.1.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor scram when one or more monitored parameters exceed their specified limits, to preserve the integrity of the fuel cladding and the Reactor Coolant System (RCS) and minimize the energy that must be absorbed following a loss of coolant accident (LOCA). This can be accomplished either automatically or manually.

The protection and monitoring functions of the RPS have been designed to ensure 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 LCOs on other reactor system parameters and equipment performance. The LSSS are defined in this Specification as the Allowable Values, which, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits, including Safety Limits (SLs) during Design Basis Accidents (DBAs).

The RPS, as described in the FSAR, Section 7.2 (Ref. 1), includes sensors, relays, bypass circuits, and switches that are necessary to cause initiation of a reactor scram. Functional diversity is provided by monitoring a wide range of dependent and independent parameters. The input parameters to the scram logic are from instrumentation that monitors reactor vessel water level, reactor vessel pressure, neutron flux, main steam line isolation valve position, turbine control valve (TCV) fast closure trip oil pressure (indicated by TCV low hydraulic pressure), turbine stop valve (TSV) position, drywell pressure, scram pilot air header pressure, and scram discharge volume (SDV) water level, as well as reactor mode switch in shutdown position, manual, and RPS channel test switch scram signals. There are at least four redundant sensor input signals from each of these parameters (with the exception of the reactor mode switch in shutdown, manual, and RPS channel test switch scram signals). Most channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay deenergizes, which then outputs an RPS trip signal to the trip logic.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux - High, Setdown (continued)

14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

The Average Power Range Monitor Neutron Flux - High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

In MODE 1, the Average Power Range Monitor Neutron Flux - High Function provides protection against reactivity transients and the RWM and rod block monitor protect against control rod withdrawal error events.

2.b. Average Power Range Monitor Flow Biased Simulated Thermal Power - High

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function monitors neutron flux to approximate the THERMAL POWER being transferred to the reactor coolant. The APRM neutron flux is electronically filtered with a time constant representative of the fuel heat transfer dynamics to generate a signal proportional to the THERMAL POWER in the reactor. The trip level is varied as a function of recirculation drive flow (i.e., at lower core flows, the setpoint is reduced proportional to the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

Alternately, if it is not desired to place the inoperable channels (or one trip system) in trip (e.g., as in the case where placing the inoperable channel or associated trip system in trip would result in a scram or RPT), Condition D must be entered and its Required Action taken.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve - Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).

For Function 8 (Turbine Stop Valve - Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip).

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1.7

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 MWD/T average core exposure Frequency is based on operating experience with LPRM sensitivity changes.

SR 3.3.1.1.8, SR 3.3.1.1.12 and SR 3.3.1.1.16

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.8 is based on the reliability analysis of Reference 9.

The 184 day Frequency of SR 3.3.1.1.16 for the scram pilot air header low pressure trip function is based on the functional reliability previously demonstrated by this function, the need for minimizing the radiation exposure associated with the functional testing of this function, and the increased risk to plant availability while the plant is in a half-scram condition during the performance of the functional testing versus the limited increase in reliability that would be obtained by the more frequent functional testing.

The 18 month Frequency of SR 3.3.1.1.12 is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.9, SR 3.3.1.1.10 and SR 3.3.1.1.13 (continued)

range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology. For MSIV-Closure, SDV Water Level-High (Float Switch), and TSV-Closure Functions, SR 3.3.1.1.13 includes physical inspection and actuation of the switches.

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from 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 performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.7). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

The Frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.10 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.1.1.11

The Average Power Range Monitor Flow Biased Simulated Thermal Power - High Function uses the recirculation loop drive flows to vary the trip setpoint. This SR ensures that the total loop drive flow signals from the flow units used to vary the setpoint are appropriately compared to a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued).

The RBM Function satisfies Criterion 3 of the NRC Policy Statement (Ref. 10).

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value to ensure that no single instrument failure can preclude a rod block from this Function. The setpoints are calibrated consistent with applicable setpoint methodology (nominal trip setpoint).

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 29% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ 1.75, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ 1.44, no RWE event will result in exceeding the MCPR

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.3.2.2 and SR 3.3.3.2.3

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

The Frequency of SR 3.3.3.2.2 is based upon the assumption of a 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The 18 month Frequency of SR 3.3.3.2.3 is based upon operating experience and consistency with the typical industry refueling cycle.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 19.

2. FSAR Section 7.18.

3. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

Table B 3.3.3.2-1 (Page 1 of 3)
Backup Control System Instrumentation and Controls

FUNCTION	NUMBER REQUIRED
Instrument Parameter	
1. Reactor Water Level Indication	1
2. Reactor Pressure Indication	1
3. Suppression Pool Temperature Indication	1
4. Suppression Pool Level Indication	1
5. Drywell Pressure Indication	1
6. RHR Flow Indication	1
7. RCIC Flow Indication	1, note a
8. RCIC Turbine Speed Indication	1
9. Drywell Temperature Indication	note o
10. RHRSW Header Pressure	1, note p
Transfer/Control Parameter	
11. Main Steam Relief Valve (MSRV) Transfer & Control	3, note b
12. Main Steam Isolation Valve (MSIV) Transfer & Control (Closure)	4, note c
13. Main Steam Drain Line Isolation Valve	1, note d
14. RHRSW Pumps	note e
15. RHRSW Discharge Valves for RHR Loop I Heat Exchangers	2, note f
note a: RCIC flow indication may be obtained from the Flow Indicating Controller	
note b: 1 required for each of 3 MSRVs.	
note c: 1 MSIV required per penetration, may be either inboard valve or outboard valve.	
note d: 1 Main Steam Drain Line isolation valve required, may be either inboard valve or outboard valve.	
note e: There are 12 RHRSW pumps. All are equipped with emergency transfer switches. 2 of the 12 must be available for EECW service (supports all units) and an additional 1 must be available for RHRSW service.	
note f: 1 Discharge Valve per RHR Loop I Heat Exchanger for a total of 2.	
note o: Drywell Temperature Indication not required for Unit 3	
note p: The RHRSW Pressure indicator for the Header of the RHRSW Pump that supports RHR service is required.	

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level - Low Low, Level 2 or Reactor Steam Dome Pressure - High setpoint is reached, the recirculation pump motor breakers trip.

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic.

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure - High and two channels of Reactor Vessel Water Level - Low Low, Level 2 in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each Function. Thus, either two Reactor Vessel Water Level - Low Low, Level 2 or two Reactor Pressure - High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective motor breakers).

There are two motor breakers provided for each of the two recirculation pumps for a total of four breakers. The output of each trip system is provided to one of the two breakers for each recirculation pump.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The individual Functions are required to be OPERABLE in MODE 1 to protect against catastrophic/multiple failures of the Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure - High and Reactor Vessel Water Level - Low Low, Level 2 Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. Reactor Vessel Water Level - Low Low, Level 2 (LS-3-58A1, LS-3-58B1, LS-3-58C1, and LS-3-58D1)

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT System is initiated at Level 2 to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level - Low Low, Level 2 (LS-3-58A1, LS-3-58B1, LS-3-58C1, and LS-3-58D1) (continued)

Four channels of Reactor Vessel Water Level - Low Low, Level 2 with two channels in each trip system, are available and required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level - Low Low, Level 2 Allowable Value is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation.

b. Reactor Steam Dome Pressure - High (PIS-3-204A, PIS-3-204B, PIS-3-204C, and PIS-3-204D)

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure - High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code limits.

The Reactor Steam Dome Pressure - High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor Steam Dome Pressure - High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure - High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code limits.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), and the Automatic Depressurization System (ADS). The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS - Operating."

Portions of the ECCS instrumentation also provide for the generation of the Common Accident Signal which initiate the DGs and EECW System. Refer to LCO 3.8.1, "AC Systems-Operating," for operability requirements of the Common Accident Signal Logic.

Core Spray System

The CS System may be initiated by automatic means. Each pump can be controlled manually by a control room remote switch. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low, Level 1 or both Drywell Pressure - High and Reactor Steam Dome Pressure - Low. Reactor water level and drywell pressure are each monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of these trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function. The Reactor Steam Dome Pressure - Low variable is monitored by two transmitters for each trip system. The outputs from these transmitters are connected to relays arranged in a one-out-of-two logic.

(continued)

BASES

BACKGROUND

Core Spray System (continued)

The high drywell pressure and low reactor water level initiation signals are sealed in signals and must be manually reset. Upon receipt of an initiation signal, if normal AC power is available, the four core spray pumps start one at a time, in order, at 0, 7, 14, and 21 seconds. If normal AC power is not available, the four core spray pumps start seven seconds after standby power becomes available. (The LPCI pumps start as soon as standby power is available.)

The CS test line isolation valve is closed on a CS initiation signal to allow full system flow assumed in the accident analyses.

The CS pump discharge flow is monitored by a flow switch. When the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow assumed in the accident analysis.

The CS System logic also receives signals from transmitters which monitor the pressure in the reactor to ensure that, before the injection valves open, the reactor pressure has fallen to a value below the CS System's maximum design pressure. Reactor pressure is monitored by four redundant transmitters, which are, in turn, connected to four trip units (two per trip system). The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two logic for each CS trip system.

Low Pressure Coolant Injection System

The LPCI is an operating mode of the Residual Heat Removal (RHR) System, with two LPCI subsystems. The LPCI subsystems may be initiated by automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low Low, Level 1 or both Drywell Pressure - High and Reactor Steam Dome Pressure - Low. Each of these diverse variables is monitored by four redundant transmitters, which, in turn, are connected to four trip

(continued)

BASES

BACKGROUND

Low Pressure Coolant Injection System (continued)

Low reactor water level in the shroud is detected by two additional instruments which inhibit the manual initiation of other modes of RHR (e.g., suppression pool cooling) when LPCI is required. Manual overrides for the inhibit logic are provided.

High Pressure Coolant Injection System

The HPCI System may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level - Low Low, Level 2 or Drywell Pressure - High. Each of these variables is monitored by four redundant transmitters, which are, in turn, connected to multiple trip units. The outputs of the trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

The HPCI pump discharge flow is monitored by a flow switch. Upon automatic initiation, when the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow, however, the flow rates assumed in the accident analysis can be achieved with the minimum flow valve open.

The HPCI test line isolation valve is closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis.

The HPCI System also monitors the water levels in the HPCI pump supply header from the condensate storage tank (CST) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in the HPCI pump supply header from the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

detect low water level in the HPCI pump supply header from the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close.

The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level - High, Level 8 trip, at which time the HPCI turbine trips, which causes the turbine's stop valve to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI System automatically restarts if a Reactor Vessel Water Level - Low Low, Level 2 signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High or ADS High Drywell Pressure Bypass Timer; confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory); and CS or LPCI Pump Discharge Pressure - High are all present and the ADS Initiation Timer has timed out. There are two transmitters each for Reactor Vessel Water Level - Low Low Low, Level 1 and Drywell Pressure - High, and one transmitter for confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory) in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that

(continued)

BASES

BACKGROUND

Automatic Depressurization System (continued)

the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the ADS Initiation Timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

The ADS also monitors the discharge pressures of the four LPCI pumps and the four CS pumps. Each ADS trip system includes two discharge pressure permissive switches from all four LPCI pumps and one discharge pressure permissive switch from all four CS pumps. The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. CS pumps (A or B and either C or D) or any one of the four LPCI pumps is sufficient to permit automatic depressurization.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from each of the following variables: Reactor Vessel Water Level - Low Low Low, Level 1; Drywell Pressure - High; High Drywell Pressure Bypass Timer; and Pump Discharge Pressure High. One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level - Low, Level 3 (confirmatory). Either the Drywell Pressure - High or the Drywell Pressure Bypass Timer contacts and all remaining contacts in both logic strings must close and the ADS initiation timer must time out to initiate an ADS trip system. Either the A or B trip system will cause all the ADS relief valves to open. Once the Drywell Pressure - High signal, the ADS High Drywell Pressure Bypass Timer, or the ADS initiation signal is present, it is individually sealed in until manually reset.

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

In general, the individual Functions are required to be OPERABLE in the MODES or other specified conditions that may require ECCS initiation to mitigate the consequences of a design basis transient or accident. To ensure reliable ECCS function, a combination of Functions is required to provide primary and secondary initiation signals.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis.

Core Spray and Low Pressure Coolant Injection Systems

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D)

Low reactor pressure vessel (RPV) water level indicates that the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. The low pressure ECCS are initiated at Level 1 to ensure that core spray and flooding functions are available to prevent or minimize fuel damage. The Reactor Vessel Water Level - Low Low Low, Level 1 is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating," for operability requirements of the Common Accident Signal Logic). The Reactor Vessel Water Level - Low Low Low, Level 1 is one of the Functions assumed to be OPERABLE and capable of initiating the ECCS during the transients analyzed in References 1 and 3. In addition, the Reactor Vessel Water Level - Low Low Low, Level 1 Function is directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the Reactor Protection System (RPS), ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level - Low Low Low, Level 1 (LS-3-58A-D) (continued)

Reactor Vessel Water Level - Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level - Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level - Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS - Shutdown," for Applicability Bases for the low pressure ECCS subsystems.

1.b, 2.b. Drywell Pressure - High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS is initiated upon receipt of the Drywell Pressure - High Function in order to minimize the possibility of fuel damage. The Drywell Pressure - High is also utilized in the development of the Common Accident Signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating" for operability requirements of the Common Accident Signal Logic). The Drywell Pressure - High Function, along with the Reactor Steam Dome Pressure - Low Function, are directly assumed in the analysis of the recirculation line break (Ref. 2). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

(continued)

BASES (continued)

ACTIONS

A Note has been provided to modify the ACTIONS related to ECCS instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are initiated by Functions 1.a, 1.b, 2.a, and 2.b (e.g., low pressure ECCS). The Required Action B.2 system would be HPCI. For Required Action B.1, redundant automatic initiation capability is lost if (a) two or more Function 1.a channels are inoperable and untripped such that both trip systems lose initiation capability, (b) two or more Function 2.a channels are inoperable and untripped such that both trip systems lose initiation capability, (c) two or more Function 1.b channels are inoperable and untripped such that both trip systems lose initiation capability, or (d) two or more Function 2.b channels are inoperable and untripped such that both trip systems lose initiation capability. For low pressure ECCS, since each inoperable

(continued)

BASES

ACTIONS

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

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that redundant features in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable, untripped channels within the same Function as described in the paragraph above. For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to inoperable, untripped channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions 1.c, 2.d, 1.e, 2.d, and 2.f (i.e., low pressure ECCS). Redundant automatic initiation capability is lost if

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

either (a) four Function 1.c channels are inoperable (i.e., both channels in both trip systems are inoperable), (b) two or more Function 2.c channels are inoperable such that both trip systems lose initiation capability, (c) two or more Function 2.d channels are inoperable such that both trip systems lose initiation capability, (d) one or more Function 1.e channels are inoperable in both trip systems (i.e., at least one CS pump in both subsystems is affected), or (e) multiple Function 2.f channels are inoperable such that the trip systems cannot start both LPCI pumps in at least one subsystem. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action C.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system to be declared inoperable. However, since channels for both low pressure ECCS subsystems are inoperable (e.g., both CS subsystems), and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in both subsystems being concurrently declared inoperable. For Functions 1.c, 2.c, 1.e, 2.d, and 2.f, the affected portions are the associated low pressure ECCS pumps. As noted (Note 1), Required Action C.1 is only applicable in MODES 1, 2, and 3.

In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of automatic initiation capability for 24 hours (as allowed by Required Action C.2) is allowed during MODES 4 and 5.

Note 2 states that Required Action C.1 is only applicable for Functions 1.c, 2.c, 1.e, 2.d, and 2.f. Required Action C.1 is also not applicable to Function 3.c (which also requires entry into this Condition if a channel in this Function is inoperable). The loss of one Function 3.c channel results in a loss of the Function (two-out-of-two logic). This loss was considered during the development of Reference 4 and considered acceptable for the 24 hours allowed by Required Action C.2.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action C.1, the Completion Time only begins upon discovery that redundant features in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, Condition H must be entered and its Required Action taken. The Required Actions do not allow placing the channel in trip since this action would either cause the initiation or it would not necessarily result in a safe state for the channel in all events.

D.1

Required Action D.1 is intended to ensure that appropriate actions are taken if an inoperable, untripped channel within the same Function results in a complete loss of automatic component initiation capability for the HPCI System. Since Table 3.3.5.1-1 only requires one channel to be OPERABLE, automatic component initiation capability is lost if the one required Function 3.d channel or the one required Function 3.e channel is inoperable and untripped. In this situation (loss of automatic suction swap), the HPCI system must be declared inoperable within 1 hour. As noted, Required Action D.1 is only applicable if the HPCI pump suction is not aligned to the suppression pool, since, if aligned, the Function is already performed.

(continued)

BASES

ACTIONS

D.1 (continued)

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

E.1 and E.2

Required Action E.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the Core Spray Pump Discharge Flow - Low Bypass Function results in redundant automatic initiation capability being lost for the feature(s). Automatic initiation capability of the Core Spray Pump Discharge Flow - Low (Bypass) Function in both CS subsystems is lost if two Function 1.d channels are inoperable. In this situation (loss of capability for both subsystems), the 7 day allowance of Required Action E.2 is not appropriate and the subsystem associated with each inoperable channel must be declared inoperable within 1 hour. Since each inoperable channel would have Required Action E.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected CS pump to be declared inoperable. However, since channels for both CS subsystems are inoperable, and the completion times started concurrently for both channels this results in all four CS pumps being concurrently declared inoperable. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to Function 1.d. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference 4 and considered acceptable for the 7 days allowed by Required Action E.2.

(continued)

BASES

ACTIONS (continued)

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 6 hours.

(continued)

BASES

ACTIONS

E.1 (continued)

The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

F.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, plant operations may continue if the affected penetration flow path(s) is isolated. Isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable channels.

For the RWCU Area Temperature - High Functions, the affected penetration flow path(s) may be considered isolated by isolating only that portion of the system in the associated room monitored by the inoperable channel. That is, if the RWCU pump room A area channel is inoperable, the pump room A area can be isolated while allowing continued RWCU operation utilizing the B RWCU pump.

Alternately, if it is not desired to isolate the affected penetration flow path(s) (e.g., as in the case where isolating the penetration flow path(s) could result in a reactor scram), Condition G must be entered and its Required Actions taken.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for plant operations personnel to isolate the affected penetration flow path(s).

G.1 and G.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, or any Required Action of Condition F is not met and the associated Completion Time has expired, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1 and H.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the SLC System is declared inoperable or the RWCU System is isolated. Since this Function is required to ensure that the SLC System performs its intended function, sufficient remedial measures are provided by declaring the SLC System inoperable or isolating the RWCU System.

The 1 hour Completion Time is acceptable because it minimizes risk while allowing sufficient time for personnel to isolate the RWCU System.

I.1 and I.2

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the associated penetration flow path(s) should be closed. However, if the shutdown cooling function is needed to provide core cooling, these Required Actions allow the penetration flow path(s) to remain unisolated provided action is immediately initiated to restore the channel to OPERABLE status or to isolate the RHR Shutdown Cooling System (i.e., provide alternate decay heat removal capabilities so the penetration flow path(s) can be isolated). Actions must continue until the channel is restored to OPERABLE status or the RHR Shutdown Cooling System is isolated.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each Primary Containment Isolation instrumentation Function are found in the SRs column of Table 3.3.6.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains trip capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 5 and 6) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the PCIVs will isolate the penetration flow path(s) when necessary.

SR 3.3.6.1.1

Performance of the CHANNEL CHECK once every 24 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 instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff 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 instrument has drifted outside its limit.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1.1 (continued)

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 the channels required by the LCO.

SR 3.3.6.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The 92 day Frequency of SR 3.3.6.1.2 is based on the reliability analysis described in References 5 and 6.

SR 3.3.6.1.3, SR 3.3.6.1.4 and SR 3.3.6.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequencies of SR 3.3.6.1.3, SR 3.3.6.1.4, and SR 3.3.6.1.5 are based on the magnitude of equipment drift in the setpoint analysis.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required isolation logic for a specific channel. The system functional testing performed on PCIVs in LCO 3.6.1.3 overlaps this Surveillance to provide complete testing of the assumed safety function. The LOGIC SYSTEM FUNCTIONAL TEST shall include a calibration of time delay relays and timers necessary for proper functioning of the logic. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

Operating experience has shown these components usually pass the Surveillance when performed at the Frequency provided.

REFERENCES

1. FSAR, Section 6.5.

2. FSAR, Chapter 14.

3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.

4. FSAR, Section 4.9.3.

5. NEDC-31677P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," July 1990.

6. NEDC-30851P-A Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.

(continued)

BASES

REFERENCES (continued)

7. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

8. FSAR, Section 5.2.

9. NRC letter from Richard J. Clark to Hugh G. Parris dated August 9, 1984, Safety Evaluation for Amendment Nos. 107, 101, and 74 to Facility Operating License Nos. DPR-33, DPR-52, and DPR-68 for Browns Ferry Nuclear Plant Units 1, 2, and 3 respectively.

BASES

ACTIONS

E.1 and E.2 (continued)

placed in operation (Required Action E.1) while the other CREV subsystem can be declared inoperable (Required Action E.2).

The 1 hour Completion Time is intended to allow the operator time to place the CREV subsystem(s) in operation. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels, for placing the associated CREV subsystem(s) in operation, or for entering the applicable Conditions and Required Actions for the inoperable CREV subsystem(s).

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each CREV System instrumentation Function are located in the SRs column of Table 3.3.7.1-1.

The Surveillances are modified by a Note (Note 2) to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains CREV System initiation capability. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. This Note is based on the reliability analysis (Refs. 3 and 4) assumption of the average time required to perform channel surveillance. That analysis demonstrated that the 6 hour testing allowance does not significantly reduce the probability that the CREV System will initiate when necessary.

The Surveillances are modified by a third Note (Note 3) to indicate that for Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of a CHANNEL CALIBRATION or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 24 hours provided the downscale trip of the inoperable channel is placed in the tripped condition. Upon completion of the Surveillance or maintenance, or expiration of the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

24 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

SR 3.3.7.1.1

Performance of the CHANNEL CHECK once every 24 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 instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, 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 instrument has drifted outside its limit.

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

SR 3.3.7.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

(continued)

B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The LOP instrumentation monitors the 4.16 kV shutdown boards. Offsite power is the preferred source of power for the 4.16 kV shutdown boards. If the monitors determine that insufficient power is available, the boards are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources.

Each 4.16 kV shutdown board has its own independent LOP instrumentation and associated trip logic. The voltage for each board is monitored at two levels, which can be considered as two different undervoltage Functions: Loss of Voltage and 4.16 kV Shutdown Board Undervoltage Degraded Voltage. Each Function causes various board transfers and disconnects.

The Degraded Voltage Function is monitored by three undervoltage relay channels for each shutdown board, whose outputs are arranged in a two-out-of-three logic configuration (Ref. 1). The channels compare measured input signals with pre-established setpoints. When the setpoint is exceeded for two-of-three degraded voltage channels, the logic energizes timers which provides a LOP trip signal to the shutdown board logic.

The Loss of Voltage Function is monitored by two undervoltage relay pairs for each shutdown board, where outputs are arranged in a two-out-of-two logic configuration (Ref. 1). The channels include four electro-mechanical relays, two of which must deenergize to start the associated diesel generator and another two which must deenergize to initiate load shed of the associated 4.16 kV shutdown board.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environmental effects (for unit channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for.

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis. The channel devices for each shutdown board are listed in Table B 3.3.8.1-1.

1. 4.16 kV Shutdown Board Undervoltage (Loss of Voltage)

Loss of voltage on a 4.16 kV shutdown board indicates that offsite power may be completely lost to the respective shutdown board and is unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the board is transferred from offsite power to DG power upon total loss of shutdown board voltage for 1.5 seconds. The transfer will not occur if the voltage recovers to the specified Allowable Value for Reset Voltage within 1.5 seconds. This ensures that adequate power will be available to the required equipment.

The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that power is available to the required equipment.

Two channels of 4.16 kV Shutdown Board Undervoltage (Loss of Voltage) Function per associated shutdown board are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1, "AC Sources - Operating," and 3.8.2, "AC Sources - Shutdown," for Applicability Bases for the DGs.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

2. 4.16 kV Shutdown Board Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV shutdown board indicates that, while offsite power may not be completely lost to the respective shutdown board, available power maybe insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the board is transferred from offsite power to onsite DG power when the voltage on the board drops below the Degraded Voltage Function Allowable Values (degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The Board Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

Three channels of 4.16 kV Shutdown Board Undervoltage (Degraded Voltage) Function per associated board are required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS

A Note has been provided to modify the ACTIONS related to LOP instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable LOP instrumentation channels provide appropriate compensatory measures for separate inoperable channels.

(continued)

BASES

ACTION (continued)

As such, a Note has been provided that allows separate Condition entry for each inoperable LOP instrumentation channel.

A.1 and A.2

With one of the degraded voltage relay channels inoperable, Required Action A.2 provides a 15 day allowable out of service time to restore the relay channel to OPERABLE status provided the other two degraded voltage relay channels and associated timers and the loss of voltage relay channel(s) on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of the other degraded voltage relay channels and associated timers and loss of voltage relay channels is therefore required (Required Action A.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition C or D, as applicable, must be entered immediately. The 15 day allowable out of service time is justified based on the two-out-of-three permissive logic scheme provided for these relays. If the inoperable relay channel cannot be restored to OPERABLE status within the allowable out of service time, the degraded voltage relay channel must be placed in the tripped condition per Required Action A.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

B.1 and B.2

With one or more loss of voltage relay channels inoperable, the Function is not capable of performing the intended function. Required Action B.2 provides a 10 day allowable out of service time provided two or more degraded voltage

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

relay channels and associated timers on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of two or more degraded voltage relay channels and associated timers is therefore required (Required Action B.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition D must be entered immediately. The 10 day allowable out of service time is justified since the degraded voltage relay channels on the same shutdown board are independent of the loss of voltage relay channels and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channels cannot be restored to OPERABLE status within the allowable out of service time, the channel(s) must be placed in the tripped condition per Required Action B.2. Placing the inoperable channel(s) in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel(s) in trip (e.g., as in the case where placing the channel(s) in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

C.1 and C.2

With two or more degraded voltage relay channels or one or more associated timers inoperable, the Function is not capable of performing the intended function. Required Action C.2 provides a 10 day allowable out of service time provided the loss of voltage relay channels on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of the loss of voltage relay channels is therefore required (Required Action C.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

equipment cannot be verified however, Condition D must be entered immediately.

The 10 day allowable out of service time is justified since the loss of voltage relay channels on the same shutdown board are independent of the degraded voltage relay channel(s) and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channel(s) cannot be restored to OPERABLE status within the allowable out of service time, the channel(s) must be placed in the tripped condition per Required Action C.2. Placing the inoperable channel(s) in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel(s) in trip (e.g., as in the case where placing the channel(s) in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

D.1 and D.2

With two or more degraded voltage relay channels or one or more associated timers and the loss of voltage relay channel(s) inoperable on the same shutdown board, the associated diesel generator will not automatically start upon degraded voltage or complete loss of voltage on that shutdown board. In this situation, Required Action D.2 provides a 5 day allowable out of service time provided the other shutdown boards and undervoltage relay channels are OPERABLE. Immediate verification of the OPERABILITY of the other shutdown boards and undervoltage relay channels is therefore required (Required Action D.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition E must be entered immediately. The 5 day allowable out of service time is justified based on the remaining redundancy of the 4.16 kV Shutdown Boards. The 4.16 kV Shutdown Boards have a

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

similar allowable out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

E.1

If any Required Action and associated Completion Time are not met, the associated Function is not capable of performing the intended function. Therefore, the associated DG(s) is declared inoperable immediately. This requires entry into applicable Conditions and Required Actions of LCO 3.8.1 and LCO 3.8.2, which provide appropriate actions for the inoperable DG(s).

SURVEILLANCE REQUIREMENTS

As noted (Note 1) at the beginning of the SRs, the SRs for each LOP instrumentation Function are located in the SRs column of Table 3.3.8.1-1.

SR 3.3.8.1.1 and SR 3.3.8.1.2

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.1.1 and SR 3.3.8.1.2 (continued)

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The Frequency is based upon the calibration interval assumed in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.1.3

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required actuation logic for a specific channel. The system functional testing performed in LCO 3.8.1 and LCO 3.8.2 overlaps this Surveillance to provide complete testing of the assumed safety functions.

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR, Figure 8.4-4.
2. FSAR, Section 6.5.
3. FSAR, Section 8.5.4.
4. FSAR, Chapter 14.
5. NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 1 CURRENT TECHNICAL SPECIFICATIONS MARKUP

Replaced 3.3.1.1 page 10 of 20 Rev. 1 with page 10 of 20 Rev. 2
Replaced 3.3.1.1 page 13 of 20 Rev. 1 with page 13 of 20 Rev. 2
Replaced 3.3.1.1 page 18 of 20 Rev. 1 with page 18 of 20 Rev. 2
Replaced 3.3.1.2 page 3 of 4 Rev. 1 with page 3 of 4 Rev. 2
Replaced 3.3.2.1 page 4 of 12 Rev. 1 with page 4 of 12 Rev. 2
Replaced 3.3.2.1 page 6 of 12 Rev. 1 with page 6 of 12 Rev. 2
Replaced 3.3.5.1 page 11 of 25 Rev. 1 with page 11 of 25 Rev. 2
Replaced 3.3.5.1 page 12 of 25 Rev. 1 with page 12 of 25 Rev. 2
Replaced 3.3.5.1 page 21 of 25 Rev. 1 with page 21 of 25 Rev. 2
Replaced 3.3.5.1 page 22 of 25 Rev. 1 with page 22 of 25 Rev. 2
Replaced 3.3.6.1 page 8 of 28 Rev. 1 with page 8 of 28 Rev. 2
Replaced 3.3.6.1 page 9 of 28 Rev. 1 with page 9 of 28 Rev. 2
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Replaced 3.3.6.1 page 16 of 28 Rev. 1 with page 16 of 28 Rev. 2
Replaced 3.3.7.1 page 3 of 16 Rev. 1 with page 3 of 16 Rev. 2

(A1) (Except as noted)

3.3.1.1-1

TABLE 3.1.A

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Required ~~Min. No. of Operable Instr. Channels Per Trip System (1)(23)~~	See Table 2 Fig.*	Trip Function	Allowable Value ~~Trip Level Setting~~ (LA2)	(L1) (3)(4) Shut-down	Applicable Modes or other specified conditions ~~Modes In Which Function Must Be Operable~~ 5(a) ~~Refuel (7)~~	Startup/ 2 ~~Hot Standby~~	1 ~~Run~~	Conditions Referenced from Required Action D.1 ~~Action (1)~~	Actions A+B
1	10	Mode Switch in Shutdown	NA (A5)	~~X~~ (L2)	X	X	X	~~1.A~~ G+H	
1	11	Manual Scram	NA (A5)	~~X~~ (L2)	X	X	X	~~1.A~~ G+H	
3	1.a	IRM ~~(16)~~ (LA2) High Flux	≤120/125 Indicated on scale	X(22)	X	X	(8) (LA2)	~~1.A~~ G+H	
3	1.b	Inoperative	NA (A5)	X(22)	X	X	(5)	~~1.A~~ G+H	
2	2.b	APRM ~~(16)(24)(25)~~ (LA2) High Flux (Flow Biased)	~~See Spec. 2.1.A.1~~ ≤ 0.58 W+62% and ≤ 120% RTP	X(21) (M8)	X(17) (LA2)	X	(15) (L6)	1.A or ~~1.B~~ F ~~(A14)~~	
2	2.c	High Flux (Fixed Trip)	≤ 120%	X(21)	X(17)	X	(15)	1.A or ~~1.B~~ F	
2	2.g	High Flux	≤ 15% rated power	X(21)	X(17)	X	(15)	~~1.A~~ G+H	
2	2.e	Inoperative (A2)	~~(13)~~ NA (A5)	Note (a) to Table 3.3.1.1				~~1.A~~ G+H	
2	2.d	Downscale	≥ 3 Indicated on Scale	(11)	(14) (LA2)	X	(12) (L6)	1.A or ~~1.B~~ F ~~(A14)~~	
2	3.	High Reactor Pressure	≤ 1055 psig	X(10) (L8)	X	X		~~1.A~~ G	
2	6.	High Drywell Pressure (14)	≤ 2.5 psig	X(8)	X(9) (A13)	X		~~1.A~~ G	
2	4.	Reactor Low Water Level (14)	≥ 538" above vessel zero	X (A12)	X	X		~~1.A~~ G	

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REV. 2
Specification 3.3.1.1

A1

~~NOTES FOR TABLE 3.1.A (Cont'd)~~

L1

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D. Scram discharge volume high level

E. APRM 15 percent scram

(see previous page)

8. Not required to be OPERABLE when primary containment integrity is not required.

A12 A13

~~9. (Deleted)~~ A1

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

LA2

13. Less than 14 OPERABLE LPRMs will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

LA2

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

LA2

M8

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first state pressure is greater than or equal to 154 psig.

LA5

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

LA2

~~20. (Deleted)~~ A1

Note (a) to Table 3.3.1.1 for Function 2a + 2c

21. Only required with any control rod withdrawn from a core cell containing one or more fuel assemblies. The APRM High Flux and Inoperative Trips do not have to be OPERABLE if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5×10^5 cps. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the Source Range Monitors.

L8

LA4

TABLE ~~4.1.B~~ 3.3.1.1-1 (A1)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Trip function ~~Instrument Channel~~ (A1)	Group (1)	Calibration	Minimum Frequency ~~(2)~~ (A3)
1,a IRM High Flux	C	Comparison to APRM on Controlled ~~Startups (6)~~ Shutdowns (M5) (L7)	Note (4) (M4) SR 3.3.1.1.6 ~~Add~~ SR 3.3.1.1.9
APRM High Flux 2,b+c Output Signal	B	Heat Balance (L4) (Proposed Note to SR 3.3.1.1.2)	Once/7 Days (A4)
2,a,b+c Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/~~Operating Cycle~~ ~~92 DAYS~~ 18 MONTHS SR 3.3.1.1.~~9~~11
2 LPRM Signal	B	TIP System Traverse (8) (LA1)	Every 1000 ~~Effective Full Power Hours~~ SR 3.3.1.1.7
3 High Reactor Pressure	A	Standard Pressure Source	Every 3 Months SR 3.3.1.1.10
6 High Drywell Pressure (LA3)	A	Standard Pressure Source	Every 3 Months
4 Reactor Low Water Level	A	Pressure Standard	Every 3 Months (A11)
7 High Water Level in Scram Discharge Volume Electronic Lvl Switches ~~(LS-85-45-A, B, G, H)~~ Float Switches ~~(LS-85-45C-F)~~	A	Calibrated Water Column (5)	Note (5)
	A	Calibrated Water Column (5)	Note (5)
5 Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
6,9 Turbine First Stage Pressure Permissive (PT-1-81A, B & PT-1-91A, B) (LA3)	B	Standard Pressure Source	Once/~~Operating Cycle~~ 18 months (A4) (9)
8 Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/~~Operating Cycle~~ 18 months (A4) SR 3.3.1.1.13
9 Turbine Stop Valve Closure	A	Note (5)	Note (5)

MWD/T average core exposure (L6) (L9)

REV. 2 ~~Revision~~ Section 3.3.1.1

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~~3.10/4.10 CORE ALTERATIONS~~

(A1)

~~LIMITING CONDITIONS FOR OPERATION~~

~~3.10.B. Core Monitoring~~

Note 1 to SR 3.3.1.2.2

LCO 3.3.1.2
Table 3.3.1.2-1

Note (a) to Table 3.3.1.2-1

(L2)

1. During CORE ALTERATIONS, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE. For an SRM (FLC) to be considered OPERABLE, the following shall be satisfied:

(LA1)

a. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major CORE ALTERATIONS in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

Table 3.3.1.2-1 Note (c)

b. Verify every 12 hours an OPERABLE SRM (FLC) is located in: (M6)

SR 3.3.1.2.2

1. The fueled region;

2. The quadrant where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region; and

3. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region.

Note: One SRM (FLC) may be used to satisfy more than one of the above.

Note 2 to SR 3.3.1.2.2

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.B. Core Monitoring~~

SR 3.3.1.2.5 1. Prior (M5) to making any CORE ALTERATIONS, the SRMs (FLCs) shall be functionally tested and checked for every 7 days

SR 3.3.1.2.1 neutron response. (M5) channel check every 12 hrs

2. Note: Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM (FLC) and no other fuel assemblies in the associated core quadrant.

Note 1 to SR 3.3.1.2.4

SR 3.3.1.2.4 ONCE PER 24 HOURS AND (M9)

Once per 12 hours during CORE ALTERATIONS, verify that the associated SRM (FLC) is reading ≥ 3 cps with a signal-to-noise ratio ≥ 3:1.

(L1) Note (b) to Table 3.3.1.2-1

~~NOTES FOR TABLE 3.2.C~~ (A1) (R1)

1. The minimum number of operable channels for each trip function is detailed for the startup and run positions of the reactor mode selector switch. The SRM, IRM, and APRM (startup mode), blocks need not be operable in "run" mode, and the APRM (flow biased) rod blocks need not be operable in "startup" mode.

   With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

3. IRM downscale is bypassed when it is on its lowest range.

4. SRMs A and C downscale functions are bypassed when IRMs A, C, E, and G are above range 2. SRMs B and D downscale function is bypassed when IRMs B, D, F, and H are above range 2.

   SRM detector not in startup position is bypassed when the count rate is ≥100 CPS or the above condition is satisfied.

Note 2 to SRs & MS

~~Note 2 to SRs A+B~~

5. During repair or calibration of equipment, not more than one SRM or RBM channel ~~not more than two APRM or IRM~~ channels may be bypassed. Bypassed channels are not counted as operable channels to meet the minimum operable channel requirements. Refer to section 3.10.B for SRM requirements during core alterations. (R1) (R1) (LA1) (A2)

6. IRM channels A, E, C, G all in range 8 or above bypasses SRM channels A and C functions.

   IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions. (R1)

7. The following operational restraints apply to the RBM only. (M3)

   Applicability

   a. ~~Both RBM channels are~~ NOT bypassed ~~when reactor power is <30 percent~~ or when a peripheral control rod is selected. INSERT A (LA1)

   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

   ACTION A+B

   c. Two RBM channels are provided and only one of these may be bypassed from the console. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour. (LA2)

   ACTION B

   d. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

INSERT A
(a) THERMAL POWER ≥ 90% RTP AND MCPR < ~~1.70~~ 1.44
(b) THERMAL POWER ≥ 29% & <90% RTP AND MCPR < ~~1.70~~ 1.75

3.3.2.1-1 (A1)

TABLE ~~4.2.C~~

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test (SR 3.3.2.1.1)		Calibration (17) (SR 3.3.2.1.4) (LA2)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/3 months	once/day (8)
APRM Downscale	(1)	(13)	once/3 months	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
1.a RBM Upscale (Flow Bias)	(1)	(13)	once/~~6 months~~ ~~184 days~~ 92 days (A5) (M4)	once/day (8)
1.c RBM Downscale	(1)	(13) (LA2)	once/~~6 months~~ 92 days	once/day (8) (R2)
1.b RBM Inoperative	(1)	(13)	N/A	once/day (8)
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
Flow Bias Comparator	(1)(15)		once/operating cycle (20)	N/A
Flow Bias Upscale	(1)(13)		once/3 months	N/A
Rod Block Logic (A3)	(R1) ~~(16)~~ (A3)		N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/operating cycle	N/A

(R1) (R1) (R1)

(M1) Proposed Function 3 SR 3.3.2.1.6

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(A1) 3.3.5.1 -1

TABLE ~~3.2.B~~ (Continued)

Minimum No. Operable Per Trip Sys(1)	Function (A2)	~~Trip Level Setting~~ Allowable Value (LA1)	Action	Remarks (LA1)
1(3)	Core Spray Loop A Discharge Pressure (PI-75-20)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	Core Spray Loop B Discharge Pressure (PI-75-48)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop A Discharge Pressure (PI-74-51)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop B Discharge Pressure (PI-74-65)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(10)	Instrument Channel - RHR Start	N/A	A	1. Starts RHR area cooler fan when respective RHR motor starts.
1(10)	Instrument Channel - Thermostat (RHR Area Cooler Fan)	≤100°F	A	1. Above trip setting starts RHR area cooler fans.
2(10)	Instrument Channel - Core Spray A or C Start	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts.
2(10)	Instrument Channel - Core Spray B or D	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts.

(LA1) (LA3) (R3) (A12) (A3)

A1

3.3.5.1-1

TABLE ~~3.2.B~~ (Continued)

Minimum No. Operable Per Trip Sys(1)	Function A2	Function	Allowable Value LA1 ~~Trip Level Setting~~	Action	Remarks LA1
1(10)		Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	≤ 100°F	A	1. Above trip setting starts Core Spray area cooler fans.
1(10)		RHR Area Cooler Fan Logic	N/A	A	
1(10)		Core Spray Area Cooler Fan Logic	N/A	A	
1(11)		Instrument Channel - Core Spray Motors A or C Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(11)		Instrument Channel - Core Spray Motors B or D Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)		Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)		Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(13)		RHRSW Initiate Logic	N/A	(14)	See Justification for Changes for BFN ISTS 3.7.2
1		RPT Logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.
1(16)	4.c 5.c	ADS Timer	t ≤ 115 sec.	~~A~~ G	1. Above trip setting in conjunction with low reactor water level permissive, low reactor water level; high drywell pressure or ADS high drywell pressure bypass timer timed out, and RHR or CSS pumps running, initiates ADS.
1(16)	4.g 5.g	ADS High Drywell Pressure Bypass Timer	t ≤ 322 sec.	~~A~~ G	1. Above trip setting, in conjunction with low reactor water level permissive, low reactor water level, ADS timer timed out and RHR or

See Justification for Changes for BFN ISTS 3.3.4.1

A9 A9 A2 LB1 LA1 R3 A3 A12 A12

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(A1) 3.3.5.1-1

TABLE ~~4.2.B~~ (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Core Spray Loop A Discharge Pressure (PI-75-20)	N/A	once/6 months	once/day
Core Spray Loop B Discharge Pressure (PI-75-48)	N/A	once/6 months	once/day
RHR Loop A Discharge Pressure (PI-74-51)	N/A	once/6 months	once/day
RHR Loop B Discharge Pressure (PI-74-65)	N/A	once/6 months	once/day
Instrument Channel - RHR Start	Tested during functional test of RHR pump (refer to Section 4.5.B) SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Thermostat (RHR Area Cooler Fan)	once/month	once/6 months	N/A
Instrument Channel - Core Spray A or C Start	Tested during functional test of core spray (refer to Section 4.5.A). SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Core Spray B or D start	Tested during functional test of core spray (refer to Section 4.5.A). SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	once/month	once/6 months	N/A

A3 R3 A3 R3 A3

3.2/4.2-48

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A1 3.3.5.1-1

TABLE ~~4.2.B~~ (Continued)
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
RHR Area Cooler Fan Logic	Tested during functional test of instrument channels, RHR motor start and thermostat (RHR area cooler fan). No other test required.	N/A	N/A
Core Spray Area Cooler Fan Logic	Tested during logic system functional test of instrument channels, core spray motor start and thermostat (core spray area cooler fan). No other test required.	N/A	N/A
Instrument Channel - Core Spray Motors A or D Start	Tested during functional test of core spray pump (refer to Section 4.5.A).	N/A	N/A
Instrument Channel Core Spray Motors B or C Start	Tested during functional test of core spray pump (refer to Section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray Loop 1 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
Instrument Channel - Core Spray Loop 2 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
RHRSW Initiate Logic	once/18 months	N/A	N/A
RPT Initiate Logic	once/month	N/A	N/A
RPT Breaker	once/operating cycle	N/A	N/A

A10 A11 R3 A12

See Justification for changes for BFN ISTS 3.4.2

See Justification for Changes for BFN ISTS 3.3.4.1

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See Justification for Changes for BFN ISTS 3.3.6.2 & 3.3.7.1

A1

3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11)	Function	Allowable Value ~~Trip Level Setting~~	Action (1)	Remarks
1(15)	Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	≤ 100 mr/hr or downscale	~~F~~	1. 1 upscale channel or 2 downscale channels will a. Initiate SGTS b. Isolate refueling floor c. Close atmosphere control system.
2(7) (8)	Instrument Channel SGTS Flow - Train A R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train B R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train C R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
1	Reactor Building Isolation Timer (refueling floor)	0 ≤ t ≤ 2 secs.	H or F	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
1	Reactor Building Isolation Timer (reactor zone)	0 ≤ t ≤ 2 secs.	G or A or H	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
2(10)	Group 1 (Initiating) Logic	N/A		1. Group 1: A Group 1 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure

SR Note 2

LA1

LA1

A9

See Justification for Changes for BFN ISTS 3.3.6.2

LA1

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A1 3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

LA1

Minimum No. Instrument Channels Operable per Trip Sys(1)(11) SR Note 2	Function	Allowable Value LA1 ~~Trip Level Setting~~	Action (1)	Remarks
1	Group 1 (Actuation) Logic	N/A	-B- D, L2	1. Group 1: A Group 1 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure
2	Group 2 (Initiating) Logic	N/A	M1, M3, L2, A or (B and E), G.2.1 & G.2.2	1. Group 2: A Group 2 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure
1	Group 2 (RHR Isolation-Actuation) Logic	N/A	-D- G.1 F	
1	Group 8 (TIP-Actuation) Logic	N/A	-J- G.1, A16, L8	
1	Group 2 (Drywell Sump Drains-Actuation) Logic	N/A	-K- G.1	
1	Group 2 (Reactor Building & Refueling Floor, and Drywell Vent and Purge-Actuation) Logic	N/A	M12, ~~F and G~~ G.1	1. Part of Group 6 Logic
2	Group 3 (Initiating) Logic	N/A	L8, ~~C F~~	1. Group 3: A Group 3 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Reactor Water Cleanup System High Temperature c. Reactor Water Cleanup System High Drain Temperature

A9, L8, M12

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A1 3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11) (SR Note 2)	Function	Allowable Value ~~Trip Level Setting~~ (LA1)	Action (1)	Remarks (LA1)
1	Group 3 (Actuation) Logic	N/A	~~G~~ F (L8)	
1	Group 6 Logic	N/A	~~F and G~~ G.1 (M12) (L8)	1. Group 6: A Group 6 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure c. Reactor Building Ventilation High Radiation
1	Group 8 (Initiating) Logic	N/A	~~B~~ G.1	1. Group 8: A Group 8 isolation is automatically actuated by only the following conditions: a. High Drywell Pressure b. Reactor Vessel Low Water Level
				2. Same as Group 2 initiating logic.
1	Reactor Building Isolation (refueling floor) Logic	N/A	H or F	
1	Reactor Building Isolation (reactor zone) Logic	N/A	H or G or A	
1(7) (8)	SGTS Train A Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train B Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train C Logic	N/A	L or (A and F)	

A9 L8 M12

See Justification for Changes for BFN ISTS 3.3.6.2

~~Refer to Table 3.2.B for RCIC and HPCI functions including Groups 4, 5, and 7 valves.~~ A1

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Proposed Note to ACTIONS (A8)

(A1)

~~NOTES FOR TABLE 3.2.A~~ (ACTION A)

(LB) within 12/24 hrs

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LCO 3.3.6.1

1. When ever the respective functions are required to be OPERABLE there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken.

Action A

Isolation capability not maintained (L1) R.A 8.1 (compl time) — Actions B & C

Action G

~~A.~~ Initiate an orderly shutdown and have the reactor in Cold Shutdown in ~~24~~ hours. (36) (L2) (12)

(A17) Mode 3 in 12 hrs (M1)

Req Act D.1

~~B.~~ Initiate an orderly load reduction and have Main Steam Lines isolated within ~~eight~~ hours. Proposed Required Actions D.2.1 + D.2.2 (L6)

Req Act. E.1 for Function 1.b (A2) (M6)

ACTION F

~~C.~~ Isolate Reactor Water Cleanup System. in 1 hour (L8)

~~D.~~ Administratively control the affected system isolation valves in the closed position within one hour ~~and then declare the affected system inoperable.~~ (A7)

E. Initiate primary containment isolation within 24 hours. (M3)

F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

G. Isolate the reactor building and start the standby gas treatment system.

H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

~~I. Deleted~~ (A1) (In one hour) (L8)

See Justification for Changes for BFN ISTS 3.3.6.2

~~J. Withdraw TIP.~~ (A9)

K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

L. If one SGTS train is inoperable take actions H or A and F. If two SGTS trains are inoperable take actions A and F.

~~2. Deleted~~ (A1)

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

(LA11) Required channels for Function 1.c (A18)

(A1) 3.3.6.1-1

TABLE ~~3.2.B~~ (Continued)

See Justification for Change to BFN ISTS Sections 3.3.5.1 + 3.3.5.2

Minimum No. Operable Per Trip Sys(1)	Function	~~Trip Level Setting~~ Allowable Value (LA1)	Action	~~Remarks~~ (LA1)
1	HPCI Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1	RCIC Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1(2)	Instrument Channel - Condensate Header Low Level (LS-73-56A & B)	≥ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
1(2)	Instrument Channel - Suppression Chamber High Level	≤ 7" above instrument zero	A	1. Above trip setting will open HPCI suction valves to the suppression chamber.
2(2)	Instrument Channel - Reactor High Water Level	≤ 583" above vessel zero	A	1. Above trip setting trips RCIC turbine.
1	4.a Instrument Channel - RCIC Turbine Steam Line High Flow	~~≤ 450" H_2O (≤ 47%) (LA1) ~~ ~~160% rated steam flow~~	~~A~~ F	1. ~~Above trip setting isolates RCIC system and trips RCIC turbine.~~ (LA1)
3(2)~~f~~ (A1) → 3(2)~~e~~	4.b Instrument Channel - RCIC Steam Supply Pressure - Low ~~(PS-71-1A-D)~~ (LA1)	≥50 psig	~~A~~ F	1. ~~Below trip setting isolates RCIC system and trips RCIC turbine.~~ (LA1)
	4.c Instrument Channel - RCIC Turbine Exhaust Diaphragm Pressure - High ~~(PS-71-11A-D)~~ (LA1)	≤20 psig	~~A~~ F	1. ~~Above trip setting isolates RCIC system and trips RCIC turbine.~~

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(A1) 3.3.6.1-1

TABLE ~~3.2.B~~ (Continued)

(See Justification for Changes for BFN ISTS 3.3.5.1 & 3.3.5.2)

Minimum No. Operable Per Trip Sys(1)	Function	Allowable Value (LA1) ~~Trip Level Setting~~	Action	Remarks (LA1)
2(2)	Instrument Channel - Reactor High Water Level	≤583" above vessel zero.	A	1. Above trip setting trips HPCI turbine.
1	3.a Instrument Channel - HPCI Turbine Steam Line High Flow	~~≤90 psi (7)~~ (LA1) ~~150% rated steam flow~~ Stub	~~A~~ F	1. Above trip setting isolates HPCI system and trips HPCI turbine.
3(2) (A1)	3.b Instrument Channel - HPCI Steam Supply Pressure - Low ~~(PS-73-1A-D)~~ (LA1)	≥100 psig	~~A~~ F	1. Below trip setting isolates HPCI system and trips HPCI turbine. (LA1)
3(2)	3.c Instrument Channel - HPCI Turbine Exhaust Diaphragm ~~(PS-73-20A-D)~~ (LA1)	≤20 psig	~~A~~ F	1. Above trip setting isolates HPCI system and trips HPCI turbine.
1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
1	RCIC System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. 2. Group 7: A Group 7 isolation is automatically actuated by only the following condition: 1. The respective turbine steam supply valve not fully closed.
(A9) 1	~~RCIC System (Isolation) Logic~~	~~N/A~~ ~~(M8) (L8) (F-8)~~		(LA1) ~~1. Includes Group 5 valves.~~ ~~2. Group 5: A Group 5 isolation is actuated by any of the following conditions:~~ ~~a. RCIC Steamline Space High Temperature~~ ~~b. RCIC Steamline High Flow~~ ~~c. RCIC Steamline Low Pressure~~ ~~d. RCIC Turbine Exhaust Diaphragm High Pressure~~
1 (16)	ADS Logic	N/A	A	

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(A1) 3.3.6.1-1

TABLE ~~3.2.B~~ (Continued) (LA1)

Minimum No. Operable Per Trip Sys(1)	Function	~~Trip Level Setting~~ Allowable Value	Action	~~Remarks~~ (LA1)
1	RHR (LPCI) System (Initiation)	N/A	B	
1	RHR (LPCI) System (Containment Cooling Spray) Logic	N/A	A	
1	HPCI System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. 2. Group 7: A Group 7 isolation is automatically actuated by only the following condition: 1. The respective turbine steam supply valve not fully closed.
1	~~HPCI System (Isolation) Logic~~	N/A		1. Includes Group 4 valves. 2. Group 4: A Group 4 isolation is actuated by any of the following conditions: a. HPCI Steamline Space High Temperature b. HPCI Steamline High Flow c. HPCI Steamline Low Pressure d. HPCI Turbine Exhaust Diaphragm High Pressure
1	Core Spray System auto initiation inhibit (Core Spray auto initiation).	N/A	B	1. Inhibit due to the core spray system of another unit. 2. The inhibit is considered the contact in the auto initiating logic only; i.e., the permissive function of the inhibit.
1	LPCI System auto initiation inhibit (LPCI auto initiation)	N/A	B	1. Inhibit due to the LPCI System of another unit. 2. The inhibit is considered the contact in the auto initiating logic only, i.e., the permissive function of the inhibit.

See Justification for Changes for BFN ISTS 3.3.5.1

A9 F B M8 L8 LA1

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~~REV. 1~~ Specification 3.3.6.1

3.3.7.1-1

A1 TABLE ~~3.2.G~~

CONTROL ROOM ISOLATION INSTRUMENTATION

LA1

Minimum # of Operable Instrument Channels per trip system	Function	Allowable Values ~~Trip Level Setting~~ LA1	Action	Remarks
Function 5. 2 1 A6	Control room air supply duct Radiation monitors (RM-90-259 A & B) LA1	210 cpm above background (4)	(2)	1. Monitors located in normal control room air supply ducts. 2. Also initiates control room emergency pressurization system.
A2 (3)	Accident signal (3)	N/A	(3)	

NOTES

Act. A ~~(1) Whenever the minimum number operable cannot be met the indicated action shall be taken.~~ A1

A1 ~~(2) Action~~ ACTION D A1

L1 Proposed Required ACTION E.1

3.2/4.2-34

Req Act D.1 One channel inoperable ~~Repair as soon as possible and~~ functionally test the other channel daily.

Req. Act D.2 Two channels inoperable LA1 ~~Repair as soon as possible~~ Functionally test the control room particulate monitor (RM-90-53) and radiation monitor (RM-90-B) once per shift. These monitors alarm in the control room on high activity. This will allow the operator to manually isolate the control room and manually initiate the emergency pressurization system. If one air supply duct radiation monitor is not operable within 30 days, declare the system initiated by these monitors inoperable and take action as specified in section 3.7.E. Req Action E.2

Req Act D.3

A2 (3) Any signal that isolates primary containment also isolates the control room and initiates the control room emergency pressurization system. These signals and the appropriate action to take if the instrumentation is unavailable is indicated in Table 3.2.A.

LA1 (4) These monitors are set to trip at 210 cpm above background, which is a radiation level corresponding to about 10^{-5} uci/cc of Xenon 133 (about 1 mrem/hr). The initial setpoint is based on manufacturers empirical formulas. This setpoint will be verified by site operating personnel.

REV. 2 ~~Rev. 1~~ 2

Specification 3.3.7.1

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BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 2 CURRENT TECHNICAL SPECIFICATIONS MARKUP

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Replaced 3.3.1.1 page 13 of 20 Rev. 1 with page 13 of 20 Rev. 2
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TABLE ~~3.1.A~~ 3.3.1.1-1 (A1) <except as noted>

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Required Min. No. of Operable Instr. Channels Per Trip System (1)(23) (A1)	Func # (See Note 2)	Trip Function	Allowable Value (LA2) ~~Trip Level Setting~~	3+4 Shut-down (L1)	Applicable MODES or other specified conditions ~~Modes in which Function Must Be Operable~~ 5/a Refuel (7)	Startup/ 2 Hot Standby	1 Run	Conditions Referenced from Required Action D.1 Action (11) (Actions A+B)
1	10	Mode Switch in Shutdown	NA (A5)	X	X	X	X	~~1.A~~ G+H
1	11	Manual Scram	NA (A5)	X	X	X	X	~~1.A~~ G+H
3	1.a	IRM ~~(16)~~ (LA2) High Flux	≤120/125 Indicated on scale	X(22)	X	(5) (LA2)		~~1.A~~ G+H
3	1.b	Inoperative	NA (A5)	X(22)	X	(8)		~~1.A~~ G+H
2	2.b	APRM ~~(16)(24)(25)~~ (LA2) High Flux (Flow Biased)	~~See Spec. 2.1.A.1~~ ≤ 0.58 W +62% and ≤ 120% RTP		X			1.A or ~~1.B~~ F (L6) (A14)
2	2.c	High Flux (Fixed Trip)	≤ 120%					1.A or ~~1.B~~ F
2	2.a	High Flux	≤ 15% rated power	X(21) (Note (a) to Table 3.3.1.1)	X(17) (M8)	X (LA2) ~~(15)~~		~~1.A~~ G+H
2	2.e	Inoperative (LA2) ~~(13)~~	NA (A5)	X(21)	X(17)	X		~~1.A~~ G+H
2	2.d	Downscale	≥ 3 Indicated on Scale	~~(11)~~	~~(11)~~ (LA2)	X ~~(12)~~		1.A or ~~1.B~~ F (L6) (A14)
2	3.	High Reactor Pressure ~~(PIS-3-22AA,BB,C,D)~~	≤ 1055 psig	X(10) (L8)	X	X		~~1.A~~ G
2	6.	High Drywell Pressure (14) ~~(PIS-64-56 A-D)~~ (LA3)	≤ 2.5 psig	X(8)	X(8) (A13)	X		~~1.A~~ G
2	4.	Reactor Low Water Level (14) ~~(LIS-3-203 A-D)~~	≥ 538" above vessel zero	X (A12)	X	X		~~1.A~~ G

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REV. 2 ~~REV. 1~~
Specification 3.3.1.1

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~~NOTES FOR TABLE 3.1.A (Cont'd)~~ A1

E. APRM 15 percent scram

F. Scram pilot air header low pressure

L1 <See previous page>

8. Not required to be OPERABLE when primary containment integrity is not required.

A12 A13

~~9. (Deleted)~~ A1

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

LA2

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

13. Less than 14 OPERABLE LPRMs will cause a trip system trip.

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

LA2

15. The APRM 15 percent scram is bypassed in the RUN Mode.

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

LA2

M8

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t).

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

LA5

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required.

LA2

~~20. (Deleted)~~ A1

Note (a) to Table 3.3.1.1 for Function 2a + 2e

21. Only required with any control rod withdrawn from a core cell containing one or more fuel assemblies. The APRM High Flux and Inoperative Trips do not have to be OPERABLE if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5 x 10^5 cps. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the Source Range Monitors.

L8

LA4

TABLE ~~4.1.B~~ 3.3.1.1-1 (A1)

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

~~Instrument Channel~~ Trip Function (A1)	Group (1)	Calibration	Minimum Frequency ~~(2)~~ (R3)
1.a IRM High Flux	C	Comparison to APRM on Controlled ~~Startups (6)~~ Shutdowns (M5) (L7)	Note (4) (M4) SR 3.3.1.1.6 Add SR 3.3.1.1.9
APRM High Flux 2.b+c Output Signal	B	Heat Balance (L4) Proposed Note to SR 3.3.1.1.2	Once/7 Days SR 3.3.1.1.2
2.a,b+c Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/~~Operating Cycle~~ ~~92 DAYS~~ 18 MONTHS (A4) SR 3.3.1.1.9-11
2 LPRM Signal	B	TIP System Traverse (8) (LA1)	Every ~~1000 Effective Power Hours~~ 1000 MWD/T Average Core Exposure (L9) Full SR 3.3.1.1.7
3 High Reactor Pressure ~~(PIS-3-22 AA, BB, C, D)~~	B	Standard Pressure Source	Once/~~6 Months~~ 184 days (A-1) (9) SR 3.3.1.1.10
6 High Drywell Pressure ~~(PIS-64-56 A-D)~~	B	Standard Pressure Source	Once/18 Months (9)
4 Reactor Low Water Level ~~(LIS-3-203 A-D)~~ (LA3)	B	Pressure Standard	Once/18 Months (9) (LA1) SR 3.3.1.1.13
7 High Water Level in Scram Discharge Volume Float Switches ~~(LS-85-45 C-F)~~	A	Calibrated Water Column	Once/18 Months
Electronic Level Switches ~~(LS-85-45 A, B, G, H)~~	B	Calibrated Water Column	Once/18 Months (9)
5 Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
8,9 Turbine First Stage Pressure Permissive ~~(PIS-1-81 A&B,~~ PIS-1-91 A&B) (LA3)	B	Standard Pressure Source	Once/18 Months (9) SR 3.3.1.1.15
8 Turbine Stop Valve Closure	A	Note (5)	Note (5)
9 Turbine Control Valve Fast Closure on Turbine Trip	A	Standard Pressure Source	Once/~~Operating Cycle~~ 18 months (A4) SR 3.3.1.1.13
13 Low Scram Pilot Air Header Pressure ~~(PS-85-35 A1, A2, B1, & B2)~~	A	Standard Pressure Source	Once/18 Months

3.1/4.1-11

REV. 2 Specification 3.3.1.1

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~~LIMITING CONDITIONS FOR OPERATION~~

~~3.10.B. Core Monitoring~~

Note 1 to SR 3.3.1.2.2

LCO 3.3.1.2 Table 3.3.1.2-1

1. During CORE ALTERATIONS, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE. For an SRM (FLC) to be considered OPERABLE, the following shall be satisfied:

Note (a) to Table 3.3.1.2-1

L2

LA1

a. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major CORE ALTERATIONS in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.)

Table 3.3.1.2-1 NOTE (c)

SR 3.3.1.2.2

b. Verify every 12 hours an OPERABLE SRM (FLC) is located in: (M6)

1. The fueled region;

2. The quadrant where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region; and

3. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region.

Note 2 to SR 3.3.1.2.2

Note: One SRM (FLC) may be used to satisfy more than one of the — above.

L1 Note (b) to Table 3.3.1.2-1

BFN Unit 2

~~SURVEILLANCE REQUIREMENTS~~

~~4.10.B. Core Monitoring~~

SR 3.3.1.2.5 1. Prior (M5) to making any CORE ALTERATIONS, the SRMs (FLCs) shall be functionally tested, and checked for neutron response.

SR 3.3.1.2.1 — every 7 days — Channel check every 12 hrs

2. Note: Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM (FLC) and no other fuel assemblies in the associated core quadrant.

Note 1 to SR 3.3.1.2.4

SR 3.3.1.2.4 ONCE PER 24 HOURS AND (M9) Once per 12 hours during CORE ALTERATIONS, verify that the associated SRM (FLC) is reading ≥ 3 cps with a signal-to-noise ratio ≥ 3:1.

~~NOTES FOR TABLE 3.2.C~~ (A1) (R1) FEB 2 4 1995

1. The minimum number of OPERABLE channels for each trip function is detailed for the STARTUP and RUN positions of the reactor mode selector switch. The SRM, IRM, and APRM (STARTUP mode), blocks need not be OPERABLE in "RUN" mode, and the APRM (flow biased) rod blocks need not be OPERABLE in "STARTUP" mode.

   With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

3. IRM downscale is bypassed when it is on its lowest range.

4. SRMs A and C downscale functions are bypassed when IRMs A, C, E, and G are above range 2. SRMs B and D downscale function is bypassed when IRMs B, D, F, and H are above range 2.

   SRM detector not in startup position is bypassed when the count rate is ≥100 CPS or the above condition is satisfied.

Note 2 to SRs & M3

5. During repair or calibration of equipment, not more than one SRM or RBM channel ~~or more than two APRM or IRM~~ channels may be bypassed. Bypassed channels are not counted as OPERABLE channels to meet the minimum OPERABLE channel requirements. ~~Refer to section 3.10.B for SRM requirements during core alterations.~~ (R1) (R1) (LA1)

6. IRM channels A, E, C, G all in range 8 or above bypasses SRM channels A and C functions.

   IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions. (R1) (A2)

7. The following operational restraints apply to the RBM only. (M3)

   Applicability:

   a. ~~Both RBM channels are bypassed when reactor power is ≥30 percent~~ NOT bypassed when a peripheral (edge) control rod is selected. INSERT A (LA1)

   b. The RBM need not be OPERABLE in the "startup" position of the reactor mode selector switch.

   Action A+B:

   c. Two RBM channels are provided and only one of these may be bypassed with the console selector. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour. (LA2)

   Action B:

   d. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

INSERT A
(a) THERMAL POWER ≥ 90% RTP AND MCPR < ~~1.44~~ 1.44
(b) THERMAL POWER ≥ 25% but < 90% RTP AND MCPR < ~~1.20~~ 1.75

33.2.1-1 (A1)

TABLE ~~4.2.C~~

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

	Function	SR 33.2.1.1 Functional Test		SR 33.2.1.4 Calibration ~~(11)~~ (LA2)	Instrument Check
	APRM Upscale (Flow Bias)	(1)	(13)	once/3 months	once/day (8)
	APRM Upscale (Startup Mode)	(1)	(13)	once/3 months	once/day (8)
(R1)	APRM Downscale	(1)	(13)	once/3 months	once/day (8)
	APRM Inoperative	(1)	(13)	N/A	once/day (8)
1.a	RBM Upscale (Flow Bias)	(1)	(13)	once/6 ~~months~~ ~~184 days~~ 92 DAYS (A5) (M4)	once/day (8)
1.c	RBM Downscale	(1)	(13) (LA2)	once/6 ~~months~~ ~~184 days~~ 92 DAYS	once/day (8) (R2)
1.b	RBM Inoperative	(1)	(13)	N/A	once/day (8)
	IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
(R1)	IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
	IRM Detector Not in Startup Position	(2) (once operating cycle)		once/operating cycle (12)	N/A
	IRM Inoperative	(1)(2)	(13)	N/A	N/A
	SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
	SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
	SRM Detector Not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
	SRM Inoperative	(1)(2)	(13)	N/A	N/A
(R1)	Flow Bias Comparator	(1)(15)		once/operating cycle (20)	N/A
	Flow Bias Upscale	(1)(15)		once/3 months	N/A
	Rod Block Logic (A3)	(R1) ~~(16)~~ (A3)		N/A	N/A
	West Scram Discharge Tank Water Level High (LS-85-46L)	once/quarter		once/18 months	N/A
(R1)	East Scram Discharge Tank Water Level High (LS-85-45M)	once/quarter		once/18 months	N/A

(M1) Proposed Function 3 SR 3.3.2.1.6

REV. 2 Specification 3.3.2.1

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3.3.5.1-1

TABLE ~~3.2.B~~ (Continued)

Minimum No. Operable Per ~~Trip Sys(1)~~	Function	Allowable Value ~~Trip Level Setting~~	Action	Remarks
1(3)	Core Spray Loop A Discharge Pressure (PI-75-20)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	Core Spray Loop B Discharge Pressure (PI-75-48)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop A Discharge Pressure (PI-74-51)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop B Discharge Pressure (PI-74-65)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(~~10~~)	Instrument Channel - RHR Start	N/A	A	1. Starts RHR area cooler fan when respective RHR motor starts.
1(10)	Instrument Channel - Thermostat (RHR Area Cooler Fan)	≤100°F	A	1. Above trip setting starts RHR area cooler fans.
2(~~10~~)	Instrument Channel - Core Spray A or C Start	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts
2(~~10~~)	Instrument Channel - Core Spray B or D	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts

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Specification 3.3.5.1

A1 3.3.5.1-1

TABLE ~~3.2.8~~ (Continued)

Minimum No. Operable Per ~~Trip Sys(1)~~ Function A2	Function	Allowable Value ~~Trip Level Setting~~ LA1	Action	Remarks LA1
1(10)	Instrument Channel - Thermostat (Core Spray ~~Area~~ Cooler Fan)	≤ 100°F	A	1. Above trip setting starts Core Spray area cooler fans.
1(10)	RHR Area Cooler Fan Logic	N/A	A	
1(10)	Core Spray Area Cooler Fan Logic	N/A	A	
1(11)	Instrument Channel - Core Spray Motors A or D Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(11)	Instrument Channel - Core Spray Motor B or C Start	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)	Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(12)	Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A1, B3, C1, and D3
1(13)	RHRSW Initiate Logic	N/A	(14)	
1	RPT Logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.

R3 A12 R3 A12

3.2/4.2-22

See Justification for Changes for BFN ISTS 3.7.2

See Justification for Changes for BFN ISTS 3.3.4.1

A1 3.3.5.1-1

TABLE ~~4.2.B~~ (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Core Spray Loop A Discharge Pressure (PI-75-20)	N/A	Once/6 months	Once/day
Core Spray Loop B Discharge Pressure (PI-75-48)	N/A	Once/6 months	Once/day
RHR Loop A Discharge Pressure (PI-74-51)	N/A	Once/6 months	Once/day
RHR Loop B Discharge Pressure (PI-74-65)	N/A	Once/6 months	Once/day
Instrument Channel - RHR Start	Tested during functional test of RHR pump (refer to section 4.5.B) SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Thermostat (RHR Area Cooler Fan)	once/month	Once/6 months	N/A
Instrument Channel - Core Spray A or C Start	Tested during functional test of core spray (refer to section 4.5.A). SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Core Spray B or D start	Tested during functional test of core spray (refer to section 4.5.A). SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	once/month	Once/6 months	N/A

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A1 3.3.5.1-1

TABLE ~~4.2.B~~ (Continued)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
RHR Area Cooler Fan Logic	Tested during functional test of instrument channels, RHR motor start and thermostat (RHR area cooler fan). No other test required.	N/A	N/A
A12 Core Spray Area Cooler Fan Logic	Tested during logic system functional test of instrument channels, core spray motor start and thermostat (core spray area cooler fan). No other test required.	N/A	N/A
A13 R3 Instrument Channel - Core Spray Motors A or D Start	Tested during functional test of core spray pump (refer to section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray Motors B or C Start	Tested during functional test of core spray pump (refer to section 4.5.A).	N/A	N/A
Instrument Channel - Core Spray Loop 1 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
A12 Instrument Channel - Core Spray Loop 2 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
RHRSW Initiate Logic	Once/18 months	N/A	N/A
RPT Initiate Logic	Once/Month	N/A	N/A
RPT Breaker	Once/operating cycle	N/A	N/A

See Justification for Changes for BFN ISTS 3.7.2

See Justification for Changes for BFN ISTS 3.3.4.1

REV. 2 ~~REV. 1~~
Specification 3.3.5.1
NOV 0 4 1991

See justification for Changes for BFN ISTS 3.3.6.2 + 3.3.7.1

A1

3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1) (11) (SR Note 2)	Function	Allowable Value (LA1) ~~Trip Level Setting~~	Action (1)	Remarks (LA1)
1(14)	Instrument Channel - Reactor Building Ventilation High Radiation - Refueling Zone	≤ 100 mr/hr or downscale	F	1. 1 upscale channel or 2 downscale channels will a. Initiate SGTS b. Isolate refueling floor c. Close atmosphere control system.
2(7) (8)	Instrument Channel SGTS Flow - Train A R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train B R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train C R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
1	Reactor Building Isolation Timer (refueling floor)	0 ≤ t ≤ 2 secs.	H or F	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
1	Reactor Building Isolation Timer (reactor zone)	0 ≤ t ≤ 2 secs.	G or A or H	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
2(10)	Group 1 (Initiating) Logic	N/A	D ~~A~~ (LS)	1. Group 1: A Group 1 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure

A9

LA1

See Justification for Changes for BFN ISTS 3.3.6.2

A1

3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)
PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11) SR Note 2	Function	~~Trip Level Setting~~ Allowable Value LA1	Action (1)	Remarks LA1
1	Group 1 (Actuation) Logic	N/A	-B- D, L2	1. Group 1: A Group 1 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure
2	Group 2 (Initiating) Logic	N/A	M1, M3, L2; A or (B and E); G.2.1 & G.2.2	1. Group 2: A Group 2 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure
1	Group 2 (RHR Isolation-Actuation) Logic	N/A	-D- G.1, (F), A16	
1	Group 8 (TIP-Actuation) Logic	N/A	L8; -J- G.1	
1	Group 2 (Drywell Sump Drains-Actuation) Logic	N/A	L8; -K- G.1	
1	Group 2 (Reactor Building & Refueling Floor, and Drywell Vent and Purge-Actuation) Logic	N/A	M12; -F-and-G- G.1	1. Part of Group 6 Logic
2	Group 3 (Initiating) Logic	N/A	L8; -G- F	1. Group 3: A Group 3 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Reactor Water Cleanup (RWCU) System High Temperature in the main steam valve vault c. RWCU System High Temperature in the RWCU pump room 2A d. RWCU System High Temperature in the RWCU pump room 2B e. RWCU System High Temperature in the RWCU heat exchanger room f. RWCU System High Temperature in the space near the pipe trench containing RWCU piping

A9, L8, M12

A1

3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

Minimum No. Instrument Channels Operable per Trip Sys(1)(11) (SL Note 2)	Function	Allowable Value ~~Trip Level Setting~~ (LA1)	Action (1)	Remarks (LA1)
1	Group 3 (Actuation) Logic	N/A	~~G~~ F (L8)	
1	Group 6 Logic	N/A	~~F and G~~ G.1 (M12)	1. Group 6: A Group 6 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure c. Reactor Building Ventilation High Radiation
1	Group 8 (Initiating) Logic	N/A	~~J~~ G.1 (L8)	1. Group 8: A Group 8 isolation is automatically actuated by only the following conditions: a. High Drywell Pressure b. Reactor Vessel Low Water Level 2. Same as Group 2 initiating logic.
1	Reactor Building Isolation (refueling floor) Logic	N/A	H or F	
1	Reactor Building Isolation (reactor zone) Logic	N/A	H or G or A	
1(7) (8)	SGTS Train A Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train B Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train C Logic	N/A	L or (A and F)	

A9 L8 M12

See Justification for Changes for BFN ISTS 3.3.6.2

~~Refer to Table 3.2.B for RCIC and HPCI functions including Groups 4, 5, and 7 valves.~~ (2)A1

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Proposed Note to Actions (A8)

~~NOTES FOR TABLE 3.2.A~~ (A1)

LCO 3.3.6.1

ACTION A

(LB1) within 12/24 hrs

Whenever the respective functions are required to be OPERABLE there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken. ACTIONS B+C

ACTION A

ACTION G

isolation capability not maintained (L1) R.A B.1 Compl Time

~~A.~~ Initiate an orderly shutdown and have the reactor in Cold Shutdown in ~~24~~ hours. (36) (L2) (12) (A17) MODE 3 in 12 hours (M1)

Req. Act. D.1

~~B.~~ Initiate an orderly load reduction and have Main Steam Lines isolated within ~~eight~~ hours. Proposed Required Actions D2.1 + D2.2 (L6) Req. Act. E.1 for Func. 1.b (A2) (M6)

ACTION F

~~C.~~ Isolate Reactor Water Cleanup System in 1 hour (L8)

~~D.~~ Administratively control the affected system isolation valves in the closed position within one hour ~~and then declare the affected system inoperable~~. (A7)

E. Initiate primary containment isolation within 24 hours. (M3)

F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

G. Isolate the reactor building and start the standby gas treatment system.

H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

~~I. Deleted~~ (A1)

~~In one hour~~ (L8) See Justification for Changes for BFN ISTS 3.3.6.2

J. ~~Withdraw TIP.~~ (A9)

K. Manually isolate the affected lines. Refer to Section 4.2.~~F~~ for the requirements of an inoperable system.

L. If one SGTS train is inoperable take actions H or A and F. If two SGTS trains are inoperable take actions A and F.

~~2. Deleted~~ (A1)

3. There ~~are~~ four sensors per steam line of which at least one sensor per trip system must be OPERABLE.

(LA1) Required channels for Function 1.c (A18)

(A1) 3.3.6.1-1
TABLE ~~3.2.B~~ (Continued)

See Justification for Change to BFN ISTS Sections 3.3.5.1 & 3.3.5.2

Minimum No. Operable Per Trip Sys(1)		Function	~~Trip Level Setting~~ Allowable Value (LA1)	Action	~~Remarks~~ (LA1)
1		HPCI Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1		RCIC Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1(2)		Instrument Channel - Condensate Header Low Level (LS-73-56A & B)	≥ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
1(2)		Instrument Channel - Suppression Chamber High Level	≤ 7" above instrument zero	A	1. Above trip setting will open HPCI suction valves to the suppression chamber.
2(2)		Instrument Channel - Reactor High Water Level (LIS-3-208A and LIS-3-208C)	≤ 583" above vessel zero	A	1. Above trip setting trips RCIC turbine.
1	4.a	Instrument Channel - RCIC Turbine Steam Line High Flow ~~(PDIS-71-1A and 1B)~~ (LA1)	~~≤ 450" H_2O (LA1)~~ ~~150% rated steam flow~~ (LA1)	~~A~~ F	1. Above trip setting isolates RCIC system and trips RCIC turbine. (LA1)
~~3(2)~~ (A1)	4.b	Instrument Channel - RCIC Steam Supply Pressure - Low ~~(PS-71-1A-D)~~ (LA1)	≥50 psig	~~A~~ F	1. Below trip setting isolates RCIC system and trips RCIC turbine. (LA1)
~~3(2)~~	4.c	Instrument Channel - RCIC Turbine Exhaust Diaphragm Pressure - High ~~(PS-71-11A-D)~~ (LA1)	≤20 psig	~~A~~ F	1. Above trip setting isolates RCIC system and trips RCIC turbine.

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A1

3.3.6.1-1
TABLE ~~3.2.B~~ (Continued)

See Justification for Changes for BFN ISTS 3.3.5.1 + 3.3.5.2

Minimum No. Operable Per Trip Sys(1)	Function	Allowable Value ~~Trip Level Setting~~ LA1	Action	Remarks LA1
2(2)	Instrument Channel - Reactor High Water Level (LIS-3-208B and LIS-3-208D)	≤583" above vessel zero.	A	1. Above trip setting trips HPCI turbine.
1	3.a Instrument Channel - HPCI Turbine Steam Line High Flow ~~(PDIS-73-1A and 1B)~~ LA1	~~≤90 psid (FT) LA1~~ ~~150% rated steam flow~~	~~A~~ F	1. Above trip setting isolates HPCI system and trips HPCI turbine.
3~~(2)~~ A1	3.b Instrument Channel - HPCI Steam Supply Pressure - Low ~~(PS-73-1A-D)~~ LA1	≥100 psig	~~A~~ F	1. Below trip setting isolates HPCI system and trips HPCI turbine.
3~~(2)~~	3.c Instrument Channel - HPCI Turbine Exhaust Diaphragm ~~(PS-73-20A-D)~~ LA1	≤20 psig	~~A~~ F	1. Above trip setting isolates HPCI system and trips HPCI turbine.
1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
1	RCIC System (Initiating) Logic	N/A	B	
A9 1	RCIC System (Isolation) Logic	N/A ~~M18 LB F B~~		1. Includes Group 5 valves. 2. Group 5: A Group 5 isolation is actuated by any of the following conditions: a. RCIC Steamline Space High Temperature b. RCIC Steamline High Flow c. RCIC Steamline Low Pressure d. RCIC Turbine Exhaust Diaphragm High Pressure LA1
1 (16)	ADS Logic	N/A	A	
1	RHR (LPCI) System (Initiation)	N/A	B	

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A1

3.3.6.1-1

TABLE ~~3.2.B~~ (Continued)

Minimum No. Operable Per Trip Sys(1)	Function	Allowable Value ~~Trip Level Setting~~ (LA1)	Action	Remarks (LA1)
1	RHR (LPCI) System (Containment Cooling Spray) Logic	N/A	A	
1.	HPCI System (Initiating) Logic	N/A	B	
1	HPCI System (Isolation) Logic	N/A		1. Includes Group 4 valves. (LA1) 2. Group 4: A Group 4 isolation is actuated by any of the following conditions: a. HPCI Steamline Space High Temperature b. HPCI Steamline High Flow c. HPCI Steamline Low Pressure d. HPCI Turbine Exhaust Diaphragm High Pressure
1	Core Spray System auto initiation inhibit (Core Spray auto initiation).	N/A	B	1. Inhibit due to the core spray system of another unit. 2. The inhibit is considered the contact in the auto initiating logic only; i.e., the permissive function of the inhibit.
1	LPCI System auto initiation inhibit (LPCI auto initiation)	N/A	B	1. Inhibit due to the LPCI System of another unit. 2. The inhibit is considered the contact in the auto initiating logic only, i.e., the permissive function of the inhibit.

A9 M8 L8 B F

See Justification for Changes for BFN ISTS 3.3.5.1

A1 3.3.7.1-1

TABLE ~~3.2.G~~

CONTROL ROOM ISOLATION INSTRUMENTATION

Minimum # of Operable Instrument Channels per trip system	Function	Allowable Value ~~Trip Level Setting~~ LA1	Action	Remarks LA1
Function 5. 2 1 A6	Control room air supply duct Radiation monitors LA1 ~~(RM-90-159 A & B)~~	270 cpm above background (4)	(2)	1. Monitors located in normal control room air supply ducts. 2. Also initiates control room emergency pressurization system. (3)
A2 (3)	Accident signal (3)	N/A		

NOTES

Act. A ~~(1) Whenever the minimum number operable cannot be met the indicated action shall be taken.~~ A1

A1 ~~(2) Actions~~ ACTION D

3.2/4.2-34

Req. Act D.1 → One channel inoperable - ~~Repair as soon as possible and~~ functionally test the other channel daily. A1

Req. Act D.2 → Two channels inoperable - ~~Repair as soon as possible~~ LA1 Functionally test the control room particulate monitor ~~(RM-90-53)~~ and radiation monitor ~~(RM-90-6)~~ once per shift. These monitors alarm in the control room on high activity. This will allow the operator to manually isolate the control room and manually initiate the emergency pressurization system. If one air supply duct radiation monitor is not operable within 30 days, declare the system initiated by these monitors inoperable and take action as specified in section 3.7.E. ← Req. Act. E.2

Req. Act. D.3 →

L1 Proposed Required ACTION E.1

A2 (3) Any signal that isolates primary containment also isolates the control room and initiates the control room emergency pressurization system. These signals and the appropriate action to take if the instrumentation is unavailable is indicated in Table 3.2.A.

LA1 (4) These monitors are set to trip at 270 cpm above background, which is a radiation level corresponding to about 10^{-5} mci/cc of Xenon-133 (about 1 mRem/hr). The initial set point is based on manufacturers empirical formulas. This setpoint will be verified by site operating personnel.

3 16

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FEB 05 1987

Specification 3.3.7.1

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

UNIT 3 CURRENT TECHNICAL SPECIFICATIONS MARKUP

Replaced 3.3.1.1 page 10 of 20 Rev. 1 with page 10 of 20 Rev. 2
Replaced 3.3.1.1 page 13 of 20 Rev. 1 with page 13 of 20 Rev. 2
Replaced 3.3.1.1 page 18 of 20 Rev. 1 with page 18 of 20 Rev. 2
Replaced 3.3.1.2 page 3 of 4 Rev.1 with page 3 of 4 Rev. 2
Replaced 3.3.2.1 page 4 of 12 Rev.1 with page 4 of 12 Rev. 2
Replaced 3.3.2.1 page 6 of 12 Rev.1 with page 6 of 12 Rev. 2
Replaced 3.3.5.1 page 10 of 24 Rev. 1 with page 10 of 24 Rev. 2
Replaced 3.3.5.1 page 11 of 24 Rev. 1 with page 11 of 24 Rev. 2
Replaced 3.3.5.1 page 20 of 24 Rev. 1 with page 20 of 24 Rev. 2
Replaced 3.3.5.1 page 21 of 24 Rev. 1 with page 21 of 24 Rev. 2
Replaced 3.3.6.1 page 9 of 31 Rev. 1 with page 9 of 31 Rev. 2
Replaced 3.3.6.1 page 10 of 31 Rev. 1 with page 10 of 31 Rev. 2
Replaced 3.3.6.1 page 11 of 31 Rev. 1 with page 11 of 31 Rev. 2
Replaced 3.3.6.1 page 14 of 31 Rev. 1 with page 14 of 31 Rev. 2
Replaced 3.3.6.1 page 17 of 31 Rev. 1 with page 17 of 31 Rev. 2
Replaced 3.3.6.1 page 18 of 31 Rev. 1 with page 18 of 31 Rev. 2
Replaced 3.3.6.1 page 19 of 31 Rev. 1 with page 19 of 31 Rev. 2
Replaced 3.3.7.1 page 3 of 16 Rev. 1 with page 3 of 16 Rev. 2

A1 <except as noted>

TABLE ~~3.1.A~~ 3.3.1.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS

Required ~~Min. No. of Operable Instr.~~ Channels Per Trip System (1)(2)	Func #	Trip Function	Allowable Value ~~Trip Level Setting~~ (LA2)	3+4 Shutdown (L1)	Applicable Modes or other specified conditions ~~Modes in Which Function Must Be Operable~~ SC(a) ~~Refuel (7)~~	Startup/2 ~~Hot Standby~~	1 ~~Run~~	Conditions Referenced from Required Action D.1 Action ~~(1)~~ Actions A+B	
1	10	Mode Switch in Shutdown	NA (A5)	* (L2)	X	X	X	~~1.A~~ G+H	
1	11	Manual Scram	NA (A5)	X	X	X	X (LA2)	~~1.A~~ G+H	
3	1.a	IRM ~~(16)~~ (LA2) High Flux	≤120/125 Indicated on scale	X(22)	X	(5)	~~1.A~~ G+H		
3	1.b	Inoperative	NA	X(22)	X	(5)	~~1.A~~ G+H		
2	2.b	APRM ~~(16)(24)(25)~~ (LA2) High Flux (Fixed Trip)	≤ 120%			X	1.A or ~~1.B~~ (L6) ~~A14~~		
2	2.c	High Flux (Flow Biased)	~~See Spec. 2.1.A.1~~ ≤ 0.58w+62% and ≤ 120% RTP			X	1.A or ~~1.B~~ F		
2	2.g	High Flux	≤ 15% rated power	X(21)	X(17)	X (15)	~~1.A~~ G+H		
2	2.e	Inoperative (LA2)	~~(13)~~ NA (A5)	X(21) (N18)	X(17) (LA2)	X	~~1.A~~ G+H		
2	2.d	Downscale	≥ 3 Indicated on Scale	Note(a) to Table 3.2.1.1 (N18)	(N18) (LA2)	X (12)	1.A or ~~1.B~~ (L6) ~~A14~~		
2	3	High Reactor Pressure ~~(PIS-3-22AA,BB,C,D)~~	≤ 1055 psig	(L8) X(10)	X	X	~~1.A~~ G		
2	6.	High Drywell Pressure (14) (LA3) ~~(PIS-64-56 A-D)~~	≤ 2.5 psig	X(8)	X(A13)	X	~~1.A~~ G		
2	4.	Reactor Low Water Level (14) ~~(LIS-3-203 A-D)~~	≥ 538" above vessel zero	X	X	X	~~1.A~~ G		

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~~NOTES FOR TABLE 3.1.A (Cont'd)~~ (A1)

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E. APRM 15 percent scram

F. Scram pilot air header low pressure (L1) <See previous page>

8. Not required to be OPERABLE when primary containment integrity is not required. (A12) (A13)

~~9. (Deleted)~~ (A1)

10. Not required to be OPERABLE when the reactor pressure vessel head is not bolted to the vessel.

(LA2)

11. The APRM downscale trip function is only active when the reactor mode switch is in RUN.

12. The APRM downscale trip is automatically bypassed when the IRM instrumentation is OPERABLE and not high.

13. Less than 14 OPERABLE LPRMs will cause a trip system trip.

(LA2)

14. Channel shared by Reactor Protection System and Primary Containment and Reactor Vessel Isolation Control System. A channel failure may be a channel failure in each system.

15. The APRM 15 percent scram is bypassed in the RUN Mode.

(LA2)

16. Channel shared by Reactor Protection System and Reactor Manual Control System (Rod Block Portion). A channel failure may be a channel failure in each system. If a channel is allowed to be inoperable per Table 3.1.A, the corresponding function in that same channel may be inoperable in the Reactor Manual Control System (Rod Block).

(M8)

17. Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

18. This function must inhibit the automatic bypassing of turbine control valve fast closure or turbine trip scram and turbine stop valve closure scram whenever turbine first stage pressure is greater than or equal to 154 psig.

(LA5)

19. Action 1.A or 1.D shall be taken only if the permissive fails in such a manner to prevent the affected RPS logic from performing its intended function. Otherwise, no action is required. (LA2)

~~20. (Deleted)~~ (A1)

21. Note (a) to Table 3.3.1.1 for function 2d + 2.e. Only required with any control rod withdrawn from a core cell containing one or more fuel assemblies. The APRM High Flux and Inoperative Trips do not have to be OPERABLE if the Source Range Monitors are connected to give a noncoincidence, High Flux scram, at 5 x 10^5 cps. The SRMs shall be OPERABLE per Specification 3.10.B.1. The removal of eight (8) shorting links is required to provide noncoincidence high-flux scram protection from the Source Range Monitors. (L8) (LA4)

A1

3.3.1.1-1

TABLE ~~4.1.B~~

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENT CALIBRATION
MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Trip Function ~~Instrument Channel~~	Group (1)	Calibration	Minimum Frequency (2)
1.a IRM High Flux	C	Comparison to APRM on Controlled ~~Startups (6)~~ Shutdowns (M5) (L7)	Note (4) SR 3.3.1.1.6 (M4) Add SR 3.3.1.1.9
2. APRM High Flux Output Signal b & c	B	Heat Balance (L4) Proposed Note to SR 3.3.1.1.2	Once Every 7 Days SR 3.3.1.1.2
2.a, b & c Flow Bias Signal	B	Calibrate Flow Bias Signal (7)	Once/~~Operating Cycle~~ (A4) ~~92 DAYS~~ 18 MONTHS SR 3.3.1.1.9-11
2 LPRM Signal	B	TIP System Traverse (8) (LA1)	Every 1000 ~~Effective Full Power Hours~~ 184 days (A4) SR 3.3.1.1.7
3 High Reactor Pressure ~~(PIS-3-22AA,BB,C,D)~~	B	Standard Pressure Source	Once/~~6 Months~~ (9) SR 3.3.1.1.10
6 High Drywell Pressure ~~(PIS-64-56-A-D)~~ (LA3)	B	Standard Pressure Source	Once/18 Months (9) MWD/T (L9)
4 Reactor Low Water Level ~~(LIS-3-203-A-D)~~	B	Pressure Standard	Once/18 Months (9) MWD/T average core exposure SR 3.3.1.1.13
7 High Water Level in Scram Discharge Volume Float Switches ~~(LS-85-45C-F)~~ Electronic Lvl Switches ~~(LS-85-45-A, B, G, H)~~	A	Calibrated Water Column (5) (LA1)	Note (5) (A4)
	B	Calibrated Water Column	Once/~~Operating Cycle~~ (9) 18 months
5 Main Steam Line Isolation Valve Closure	A	Note (5)	Note (5)
8,9 Turbine First Stage Pressure Permissive (PIS-1-81A&B, PIS-1-91A&B) (LA3)	B	Standard Pressure Source	Once/18 Months (9) SR3.3.1.1.15
9 Turbine Control Valve Fast Closure or Turbine Trip	A	Standard Pressure Source	Once/~~Operating Cycle~~ (A4) 18 months SR 3.3.1.1.13
8 Turbine Stop Valve Closure	A	Note (5)	Note (5)
13 Low Scram Pilot Air Header Pressure ~~(PS-85-35-A1, A2, B1 and B2)~~	A	Standard Pressure Source	Once/18 Months

A3

LA2

LA3

3.1/4.1-10

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REV. 2 ~~REV. 1~~
Specification 3.3.1.1

~~LIMITING CONDITIONS FOR OPERATION~~ | ~~SURVEILLANCE REQUIREMENTS~~

~~3.10.B. Core Monitoring~~ (Note 1 to SR 3.3.1.2.3)

LCO 3.3.1.2 Table 3.3.1.2-1

1. During CORE ALTERATIONS, except as specified in 3.10.B.2, two SRMs (FLCs) shall be OPERABLE. ~~For an SRM (FLC) to be considered OPERABLE, the following shall be satisfied:~~

Note (a) to Table 3.3.1.2-1 (L2)

a. ~~The SRM shall be inserted to the normal operating level.~~ (A1) (Use of special moveable, dunking type detectors during initial fuel loading and major CORE ALTERATIONS in place of normal detectors is permissible as long as the detector is connected to the normal SRM circuit.) (M6)

Table 3.3.1.2-1 Note (c)

b. Verify (every 12 hours) an OPERABLE SRM (FLC) is located in:

SR 3.3.1.2.2

1. The fueled region;

2. The quadrant where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region; and

3. A core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM (FLC) is included in the fueled region.

Note: One SRM (FLC) may be used to satisfy more than one of the above.

Note 2 to SR 3.3.1.2.2

4.10.B. Core Monitoring

SR 3.3.1.2.5 1. Prior (M5) to making any CORE ALTERATIONS, the SRMs (FLCs) shall be functionally tested (every 7 days) and checked for neutron response. (channel check every 12 hrs)

SR 3.3.1.2.1

2. Note: Not required to be met with less than or equal to four fuel assemblies adjacent to the SRM (FLC) and no other fuel assemblies in the associated core quadrant.

Note 1 to SR 3.3.1.2.4

SR 3.3.1.2.4 (ONCE PER 24 HOURS AND) Once per 12 hours during CORE ALTERATIONS, verify that the associated SRM (FLC) is reading ≥ 3 cps with a signal-to-noise ratio ≥ 3:1. (M9)

~~NOTES FOR TABLE 3.2.G~~

1. The minimum number of operable channels for each trip function is detailed for the startup and run positions of the reactor mode selector switch. The SRM, IRM, and APRM (startup mode) blocks need not be operable in "run" mode, and the APRM (flow biased) rod blocks need not be operable in "startup" mode.

   With the number of OPERABLE channels less than required by the minimum OPERABLE channels per trip function requirement, place at least one inoperable channel in the tripped condition within one hour.

2. The trip level setting shall be as specified in the CORE OPERATING LIMITS REPORT.

3. IRM downscale is bypassed when it is on its lowest range.

4. SRMs A and C downscale functions are bypassed when IRMs A, C, E, and G are above range 2. SRMs B and D downscale function is bypassed when IRMs B, D, F, and H are above range 2.

   SRM detector not in startup position is bypassed when the count rate is ≥100 counts per second or the above condition is satisfied.

Note 2 to SRs & M5

Action A&B

5. During repair or calibration of equipment, not more than one SRM or RBM channel ~~nor more than two APRM or IRM~~ channels may be bypassed. Bypassed channels are not counted as operable channels to meet the minimum operable channel requirements. Refer to section 3.10.B for SRM requirements during core alterations.

6. IRM channels A, E, C, G all in range 8 or above bypasses SRM channels A and C functions.

   IRM channels B, F, D, H all in range 8 or above bypasses SRM channels B and D functions.

7. The following operational restraints apply to the RBM only.

   Applicability

   a. ~~Both RBM channels are bypassed when reactor power is 30 percent~~ Not bypassed INSERT A2 or when a peripheral control rod is selected.

   b. The RBM need not be operable in the "startup" position of the reactor mode selector switch.

   ACTION A+B

   c. ~~Two~~ RBM channels are provided and only one of these may be bypassed from the console. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour.

   ACTION B

   d. With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

INSERT A
(a) THERMAL POWER ≥ 90% RTP AND MCPR < ~~1.44~~ 1.44
(b) THERMAL POWER ≥ 29% & < 90% RTP AND MCPR < ~~1.75~~ 1.75

3.3.2.1-1 (A1)

TABLE ~~4.2.C~~

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE ROD BLOCKS

Function	Functional Test (SR 3.3.2.1.1)		Calibration ~~(17)~~ (SR 3.3.2.1.4) (LA2)	Instrument Check
APRM Upscale (Flow Bias)	(1)	(13)	once/3 months	once/day (8)
APRM Upscale (Startup Mode)	(1)	(13)	once/3 months	once/day (8)
APRM Downscale	(1)	(13)	once/3 months	once/day (8)
APRM Inoperative	(1)	(13)	N/A	once/day (8)
1.a RBM Upscale (Flow Bias)	(1)	(13)	once/~~6 months~~ ~~184 days~~ 92 DAYS (A5) (M4)	once/day (8)
1.c RBM Downscale	(1)	(13)	once/~~6 months~~ ~~184 days~~ 92 DAYS (A5) (M4)	once/day (8)
1.b RBM Inoperative	(1)	(13)	N/A	once/day (8)
IRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
IRM Detector Not in Startup Position	(2) (once operating cycle)		once/operating cycle (12)	N/A
IRM Inoperative	(1)(2)	(13)	N/A	N/A
SRM Upscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Downscale	(1)(2)	(13)	once/3 months	once/day (8)
SRM Detector Not in Startup Position	(2) (once/operating cycle)		once/operating cycle (12)	N/A
SRM Inoperative	(1)(2)	(13)	N/A	N/A
Flow Bias Comparator	(1)(15)		once/operating cycle (20)	N/A
Flow Bias Upscale	(1)(15)		once/3 months	N/A
Rod Block Logic (A3)	(R1)	~~(16)~~ (A3)	N/A	N/A
West Scram Discharge Tank Water Level High (LS-85-45L)	once/quarter		once/operating cycle	N/A
East Scram Discharge Tank Water Level High (LS-85-45H)	once/quarter		once/operating cycle	N/A

(R1) (R1) (R1) (R1) (M1) Proposed Function 3 SR 3.3.2.1.6

(LA2) (R2)

REV. 2 ~~REV. 1~~ Specification 3.3.2.1

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A1 3.3.5.1-1

TABLE ~~3.2.B~~ (Continued)

Minimum No. Operable Per Function (A2) ~~Trip Sys(1)~~	Function	Allowable Value ~~Trip Level Setting~~	Action	Remarks (LA1)
1	RHR (LPCI) System (Initiation)	N/A	B	
1	RHR (LPCI) System (Containment Cooling Spray) Logic	N/A	A	
1	HPCI System (Initiating) Logic	N/A	B	
1	HPCI System (Isolation) Logic	N/A	B	
1(3)	Core Spray Loop A Discharge Pressure (PI-75-20)	0 - 500 psig Indicator (9)	D	
1(3)	Core Spray Loop B Discharge Pressure (PI-75-48)	0 - 500 psig Indicator (9)	D	
1(3)	RHR Loop A Discharge Pressure (PI-74-51)	0 - 450 psig Indicator (9)	D	
1(3)	RHR Loop B Discharge Pressure (PI-74-65)	0 - 450 psig Indicator (9)	D	
1(10)	Instrument Channel - RHR Start	N/A	A	

A12

1. Includes Group 7 valves.
2. Group 7: A Group 7 isolation is automatically actuated by only the following condition:
   1. The respective turbine steam supply valve not fully closed.

1. Includes Group 4 valves.
2. Group 4: A Group 4 isolation is actuated by any of the following conditions:
   a. HPCI Steamline Space High Temperature
   b. HPCI Steamline High Flow
   c. HPCI Steamline Low Pressure
   d. HPCI Turbine Exhaust Diaphragm High Pressure

See Justification for changes for BFN ISTS 3.3.6.1

LA3 R3

1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1. Part of filled discharge pipe requirements. Refer to Section 4.5.

1. Starts RHR area cooler fan when respective RHR motor starts.

A12 R3 LA1

REV. 2 ~~REV. 1~~ Specification 3.3.5.1

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(A1) 3.3.5.1-1

TABLE ~~3.2.B~~ (Continued)

Minimum No. Operable Per Function (A2) ~~Trip Sys(1)~~	Function	Allowable (LA1) ~~Trip Level Setting~~	Action	Remarks (LA1)
(LA3) 1(10)	Instrument Channel - Thermostat (RHR Area Cooler Fan)	≤100°F	A	1. Above trip setting starts RHR area cooler fans.
(LA3) 2(10)	Instrument Channel - Core Spray A or C Start	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts.
(A12) (R3) 2(10)	Instrument Channel - Core Spray B or D	N/A	A	1. Starts Core Spray area cooler fan when Core Spray motor starts.
(LA3) 1(10)	Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	≤ 100°F	A	1. Above trip setting starts Core Spray area cooler fans.
(A12) (R3) 1(10)	RHR Area Cooler Fan Logic	N/A	A	
1(10)	Core Spray Area Cooler Fan Logic	N/A	A	
1(11)	Instrument Channel - Core Spray Motors A or C Start	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
1(11)	Instrument Channel - Core Spray Motors B or D Start	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
(A12) 1(12)	Instrument Channel - Core Spray Loop 1 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
1(12)	Instrument Channel - Core Spray Loop 2 Accident Signal (15)	N/A	A	1. Starts RHRSW pumps A3, B1, C3, and D1
1	RPT Logic	N/A	(17)	1. Trips recirculation pumps on turbine control valve fast closure or stop valve closure > 30% power.
1(13)	RHRSW Initiate Logic	N/A	(14)	

(LA4) (R3)

See Justification for changes for BFN ISTS 3.7.2

See Justification for Changes for BFN ISTS 3.3.4.1

A1 3.3.5.1-1

TABLE ~~4.2.B~~ (Cont'd)
SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
Core Spray Loop A Discharge Pressure (PI-75-20)	N/A	once/6 months	once/day
Core Spray Loop B Discharge Pressure (PI-75-48)	N/A	once/6 months	once/day
RHR Loop A Discharge Pressure (PI-74-51)	N/A	once/6 months	once/day
RHR Loop B Discharge Pressure (PI-74-65)	N/A	once/6 months	once/day
Instrument Channel - RHR Start	Tested during functional test of RHR pump (refer to Section 4.5.B) SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Thermostat (RHR Area Cooler Fan)	once/month	once/6 months	N/A
Instrument Channel - Core Spray A or C Start	Tested during functional test of core spray (refer to Section 4.5.A). SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Core Spray B or D start	Tested during functional test of core spray (refer to Section 4.5.A). SR 3.3.5.1.6	N/A	N/A
Instrument Channel - Thermostat (Core Spray Area Cooler Fan)	once/month	once/6 months	N/A

R3 A3 A1 R3 A3 A1 A1 A1 A3 R3

REV. 2 ~~REV. 3~~ Specification 3.3.5.1

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A1 3.3.5.1-1

TABLE ~~4.3.8~~ (Cont'd)

SURVEILLANCE REQUIREMENTS FOR INSTRUMENTATION THAT INITIATE OR CONTROL THE CSCS

Function	Functional Test	Calibration	Instrument Check
RHR Area Cooler Fan Logic	Tested during functional test of instrument channels, RHR motor start and thermostat (RHR area cooler fan). No other test required.	N/A	N/A
A12 Core Spray Area Cooler Fan Logic	Tested during logic system functional test of instrument channels, core spray motor start and thermostat (core spray area cooler fan). No other test required.	N/A	N/A
~~A13~~ R3 Instrument Channel - Core Spray Motors A or D Start	Tested during functional test of core spray pump (refer to Section 4.5.A).	N/A	N/A
Instrument Channel Core Spray Motors B or C Start	Tested during functional test of core spray pump (refer to Section 4.5.A).	N/A	N/A
RPT Initiate Logic	once/month	N/A	N/A
RPT Breaker	once/operating cycle	N/A	N/A
~~A12~~ Instrument Channel - Core Spray Loop 1 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
A12 Instrument Channel - Core Spray Loop 2 Accident Signal	Tested during logic system functional test of core spray system.	N/A	N/A
RHRSW Initiate Logic	once/18 months	N/A	N/A

See justification for changes for BFN ISTS 3.7.2

See justification for changes for BFN ISTS 3.3.4.1

See Justification for Changes for BFN ISTS 3.3.6.2

A1 . 3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

LA1

Minimum No. Instrument Channels Operable per Trip Sys(1)(11) SR Note 2	Function	Allowable Value ~~Trip Level Setting~~ LA1	Action (1)	Remarks
2(7) (8)	Instrument Channel SGTS Flow - Train B R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
2(7) (8)	Instrument Channel SGTS Flow - Train C R. H. Heaters	≥2000 cfm and ≤ 4000 cfm	H and (A or F)	Below 2000 cfm airflow R.H. heaters shall be shut off.
1	Reactor Building Isolation Timer (refueling floor)	0 ≤ t ≤ 2 secs.	H or F	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
1	Reactor Building Isolation Timer (reactor zone)	0 ≤ t ≤ 2 secs.	G or A or H	1. Below trip setting prevents spurious trips and system perturbations from initiating isolation.
2(10) A9	Group 1 (Initiating) Logic	N/A	L3 D-A	1. A Group 1 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main steamline high radiation c. Main steamline high flow d. Main steamline space high temperature e. Main steamline low pressure
1 A9 K8	Group 1 (Actuation) Logic	N/A	L2 D-B	1. Group 1: A Group 1 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Low Water Level b. Main Steamline High Radiation c. Main Steamline High Flow d. Main Steamline Space High Temperature e. Main Steamline Low Pressure

LA1

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A1 3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

LA1

Minimum No. Instrument Channels Operable per Trip Sys(1) (11) (SR NOTE 2)	Function	~~Trip Level Setting~~ Allowable Value (LA1)	Action (1)	Remarks
2	Group 2 (Initiating) Logic	N/A	M1, M3, L2 → ~~A or (B and E)~~, G.2.1 & G.2.2	1. Group 2: A Group 2 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure
1	Group 2 (RHR Isolation-Actuation) Logic	N/A	~~D~~ G.1, ~~E~~	
1	Group 8 (TIP-Actuation) Logic	N/A	~~J~~ G.1, A16	
1	Group 2 (Drywell Sump Drains-Actuation) Logic	N/A	L8 → ~~K~~ G.1	
1	Group 2 (Reactor Building & Refueling Floor, and Drywell Vent and Purge-Actuation) Logic	N/A	M12 → ~~F and G~~ G.1	1. Part of Group 6 Logic
2	Group 3 (Initiating) Logic	N/A	L8 → ~~G F~~	1. Group 3: A Group 3 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. Reactor Water Cleanup (RWCU) System High Temperature in the main steam valve vault c. RWCU System High Temperature in the RWCU pump room 3A d. RWCU System High Temperature in the RWCU pump room 3B e. RWCU System High Temperature in the RWCU heat exchanger room f. RWCU System High Temperature in the space near the pipe trench containing RWCU piping

A9 L8 M12

LA1

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A1

3.3.6.1-1

TABLE ~~3.2.A~~ (Continued)

PRIMARY CONTAINMENT AND REACTOR BUILDING ISOLATION INSTRUMENTATION

LA1

Minimum No. Instrument Channels Operable per Trip Sys(1)(11) (SR Note 2)	Function	~~Trip Level Setting~~ Allowable Value (LA1)	Action (1)	Remarks
1	Group 3 (Actuation) Logic	N/A	~~G~~ F (L8)	
1	Group 6 Logic	N/A	~~F and G~~ G.1 (M12)	1. Group 6: A Group 6 isolation is actuated by any of the following conditions: a. Reactor Vessel Low Water Level b. High Drywell Pressure c. Reactor Building Ventilation High Radiation
1	Group 8 (Initiating) Logic	N/A	~~J~~ G.1 (L8)	1. Group 8: A Group 8 isolation is automatically actuated by only the following conditions: a. High Drywell Pressure b. Reactor Vessel Low Water Level 2. Same as Group 2 initiating logic.
1	Reactor Building Isolation (refueling floor) Logic	N/A	H or F	
1	Reactor Building Isolation (reactor zone) Logic	N/A	H or G or A	
1(7) (8)	SGTS Train A Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train B Logic	N/A	L or (A and F)	
1(7) (8)	SGTS Train C Logic	N/A	L or ~~(A and F)~~	

A9 L8 M12

See Justification for Changes for BFN ISTS 3.3.6.2

~~Refer to Table 3.2.B for RCIC and HPCI functions including Groups 4, 5, and 7 valves.~~ (DA1)

REV 2 ~~REV 1~~ Specification 3.3.6.1

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Proposed Note to ACTIONS (A8)

(A1) ~~NOTES FOR TABLE 3.2.A~~ ACTION A (LB1) within 12/24 hrs AUG 2 3 1991

LCO 3.3.6.1

1. Whenever the respective functions are required to be OPERABLE, there shall be two OPERABLE or tripped trip systems for each function. If the first column cannot be met for one of the trip systems, that trip system or logic for that function shall be tripped (or the appropriate action listed below shall be taken). If the column cannot be met for all trip systems, the appropriate action listed below shall be taken. ACTION A — ACTIONS B & C

ACTION G — Isolation capability not maintained (L1) → R.A. and time

~~A.~~ Initiate an orderly shutdown and have the reactor in COLD SHUTDOWN CONDITION in ~~24~~ hours. (3) (L2) (12) (A17) MODE 3 in 12 hours (M1)

Req Act D.1

~~B.~~ Initiate an orderly load reduction and have main steam lines isolated within ~~eight~~ hours. Proposed Required Actions D.2.1 + D.2.2 (L6) Req. Act E.1 for Funct 1.b (A2) (M6)

ACTION F → Isolate Reactor Water Cleanup System. in 1 hour (L8)

~~D.~~ Administratively control the affected system isolation valves in the closed position within one hour ~~and then declare the affected system inoperable~~. (A7)

E. Initiate primary containment isolation within 24 hours. (M3)

F. The handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited.

G. Isolate the reactor building and start the standby gas treatment system.

H. Immediately perform a logic system functional test on the logic in the other trip systems and daily thereafter not to exceed 7 days.

~~I. DELETED~~ (A1) ~~(A1 one hour)~~ See Justification for Changes for BFN ISTS 3.3.6.2

J. Withdraw TIP. (A9) (L8)

K. Manually isolate the affected lines. Refer to Section 4.2.E for the requirements of an inoperable system.

L. If one SGTS train is inoperable take action H or actions A and F. If two SGTS trains are inoperable take actions A and F.

~~2. Deleted~~ (A1)

3. There are four sensors per steam line of which at least one sensor per trip system must be OPERABLE. (LA1) Required channels for Function 1.c (A18)

(A1) 3.3.6.1-1

See Justification for Change to BFN ISTS Section 3.3.5.1 + 3.3.5.2

TABLE ~~3.2.B~~ (Continued)

Minimum No. Operable Per Trip Sys(1)		Function	~~Trip Level Setting~~ Allowable Value (LA1)	Action	Remarks
1		HPCI Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1		RCIC Trip System bus power monitor	N/A	C	1. Monitors availability of power to logic systems.
1(2)		Instrument Channel - Condensate Header Low Level (LS-73-56A & B)	≥ Elev. 551'	A	1. Below trip setting will open HPCI suction valves to the suppression chamber.
2(2)		Instrument Channel - Suppression Chamber High Level	≤ 7" above instrument zero	A	1. Above trip setting will open HPCI suction valves to the suppression chamber.
2(2)		Instrument Channel - Reactor High Water Level (LIS-3-208A and LIS-3-208C)	≤ 583" above vessel zero	A	1. Above trip setting trips RCIC turbine.
1	4.a	Instrument Channel - RCIC Turbine Steam Line High Flow ~~(PDIS-71-1A and 1B)~~ (LA1)	Stet ~~≤ 450" H_2O (LA1)~~ ~~150% rated steam flow~~	~~A~~ F	1. Above trip setting isolates RCIC system and trips RCIC turbine. (LA1)
3(~~2~~) (A1)	4.b	Instrument Channel - RCIC Steam Supply Pressure - ~~Low (PS-71-1A-D)~~ (LA1)	≥50 psig	~~A~~ F	1. Below trip setting isolates RCIC system and trips RCIC turbine. (LA1)
3(~~2~~) (A1)	4.c	Instrument Channel - RCIC Turbine Exhaust Diaphragm Pressure - High ~~(PS-71-11A-D)~~ (LA1)	≤ 20 psig	~~A~~ F	1. Above trip setting isolates RCIC system and trips RCIC turbine.

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(A1) 3.6.1-1

TABLE ~~3.2.B~~ (Continued)

See Justification for Changes for BFN ISTS 3.3.5.1 & 3.3.5.2

Minimum No. Operable Per Trip Sys(1)	Function	Allowable Value (LA1) ~~Trip Level Setting~~	Action	Remarks (LA1)
2(2)	Instrument Channel - Reactor High Water Level (LIS-3-208B and LIS-3-208D)	≤583" above vessel zero.	A	1. Above trip setting trips HPCI turbine.
1	3.a Instrument Channel - HPCI Turbine Steam Line High Flow ~~(PDIS-73-1A and 1B)~~ (LA1)	Step 6 ~~≤90 psi (173%) 160% rated steam flow~~ (LA1)	~~A~~ F	1. Above trip setting isolates HPCI system and trips HPCI turbine. (LA1)
3(~~2~~) (A1)	3.b Instrument Channel - HPCI Steam Supply Pressure - Low ~~(PS-73-1A-D)~~ (LA1)	≥100 psig	~~A~~ F	1. Below trip setting isolates HPCI system and trips HPCI turbine.
3(~~2~~)	3.c Instrument Channel - HPCI Turbine Exhaust Diaphragm ~~(PS-73-20A-D)~~ (LA1)	≤20 psig	~~A~~ F	1. Above trip setting isolates HPCI system and trips HPCI turbine.
1	Core Spray System Logic	N/A	B	1. Includes testing auto initiation inhibit to Core Spray Systems in other units.
1	RCIC System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. 2. Group 7: A Group 7 isolation is automatically actuated by only the following condition: 1. The respective turbine steam supply valve not fully closed.
(A9) 1	~~RCIC System (Isolation) Logic~~	~~N/A~~ ~~(MS) (LB) F B~~		1. Includes Group 5 valves. 2. Group 5: A Group 5 isolation is actuated by any of the following conditions: a. RCIC Steamline Space High Temperature b. RCIC Steamline High Flow c. RCIC Steamline Low Pressure d. RCIC Turbine Exhaust Diaphragm High Pressure (LA1)
1 (16)	ADS Logic	N/A	A	

A1 3.3.6.1-1

TABLE ~~3.2.B~~ (Continued)

LA1

Minimum No. Operable Per Trip Sys(1)	Function	Allowable Value ~~Trip Level Setting~~ (LA1)	Action	Remarks
1	RHR (LPCI) System (Initiation)	N/A	B	
1	RHR (LPCI) System (Containment Cooling Spray) Logic	N/A	A	
1	HPCI System (Initiating) Logic	N/A	B	1. Includes Group 7 valves. 2. Group 7: A Group 7 isolation is automatically actuated by only the following condition: 1. The respective turbine steam supply valve not fully closed.
1	~~HPCI System (Isolation) Logic~~	~~N/A~~ ~~M8 L8 → B F~~ A9		1. Includes Group 4 valves. 2. Group 4: A Group 4 isolation is actuated by any of the following conditions: a. HPCI Steamline Space High Temperature b. HPCI Steamline High Flow c. HPCI Steamline Low Pressure d. HPCI Turbine Exhaust Diaphragm High Pressure
	See Justification for Changes for BFN ISTS 3.3.5.1			
1(3)	Core Spray Loop A Discharge Pressure (PI-75-20)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	Core Spray Loop B Discharge Pressure (PI-75-48)	0 - 500 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop A Discharge Pressure (PI-74-51)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(3)	RHR Loop B Discharge Pressure (PI-74-65)	0 - 450 psig Indicator (9)	D	1. Part of filled discharge pipe requirements. Refer to Section 4.5.
1(10)	Instrument Channel - RHR Start	N/A	A	1. Starts RHR area cooler fan when respective RHR motor starts.

LA1

REV. 2 REV. 1 Specification 3.3.6.1

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A1 3.3.7.1-1

TABLE ~~3.2.G~~

CONTROL ROOM ISOLATION INSTRUMENTATION

Minimum # of Operable Instrument Channels per trip system	Function	Allowable Values ~~Trip Level Setting~~ LA1	Action	Remarks LA1
Function 5 ~~2~~ 1 A6	Control room air supply duct Radiation monitors LA1 ~~(RM-90-259 A & B)~~	270 cpm above background ~~(4)~~	(2)	1. Monitors located in normal control room air supply ducts. 2. Also initiates control room emergency pressurization system.
A2 (3)	Accident signal (3)	N/A	(3)	

NOTES

ACTION A ~~(1) Whenever the minimum number operable cannot be met the indicated action shall be taken.~~ A1

A1 ~~(2) Action~~ ACTION D

Req Act D.1 → One channel inoperable - Repair as soon as possible and functionally test the other channel daily.

Req Act D.2 → Two channels inoperable - Repair as soon as possible. Functionally test the control room particulate monitor ~~(RM-90-52)~~ and radiation monitor ~~(RM-90-8)~~ once per shift. These monitors alarm in the control room on high activity. This will allow the operator to manually isolate the control room and manually initiate the emergency pressurization system. If one air supply duct radiation monitor is not operable within 30 days, declare the system initiated by these monitors inoperable and take action as specified in Section 3.7.E. Req Act E.2

Req Act D.3

L1 Proposed Required ACTION E.1

(3) Any signal that isolates primary containment also isolates the control room and initiates the control room emergency pressurization system. These signals and the appropriate action to take if the instrumentation is unavailable is indicated in Table 3.2.A. A2

(4) These monitors are set to trip at 270 cpm above background, which is a radiation level corresponding to about 10^{-5} ci/cc of Xenon-133 (about 1 mRem/hr). The initial setpoint is based on manufacturers empirical formulas. This setpoint will be verified by site operating personnel. LA1

REV. 2 ~~REV. 1~~ 2 Rectification 3.3.7.1

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BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

CURRENT TECHNICAL SPECIFICATIONS JUSTIFICATION FOR CHANGES (Revised pages marked Revision 2)

Replaced ITS 3.3.1.1 pages 1 through 12 Revision 1 with ITS 3.3.1.1 pages 1 through 12 Revision 2
Replaced ITS 3.3.1.2 pages 1 through 4 Revision 1 with ITS 3.3.1.2 pages 1 through 4 Revision 2
Replaced ITS 3.3.2.1 pages 1 through 8 Revision 1 with ITS 3.3.2.1 pages 1 through 8 Revision 2
Replaced ITS 3.3.3.1 pages 1 through 6 Revision 1 with ITS 3.3.3.1 pages 1 through 6 Revision 2
Replaced ITS 3.3.4.1 pages 1 through 4 Revision 1 with ITS 3.3.4.1 pages 1 through 4 Revision 2
Replaced ITS 3.3.5.1 pages 1 through 7 Revision 1 with ITS 3.3.5.1 pages 1 through 7 Revision 2
Replaced ITS 3.3.6.1 pages 1 through 11 Revision 1 with ITS 3.3.6.1 pages 1 through 11 Revision 2
Replaced ITS 3.3.6.2 pages 1 through 8 Revision 1 with ITS 3.3.6.2 pages 1 through 8 Revision 2
Replaced ITS 3.3.7.1 pages 1 through 5 Revision 1 with ITS 3.3.7.1 pages 1 through 5 Revision 2
Replaced ITS 3.3.8.1 pages 1 through 3 Revision 1 with ITS 3.3.8.1 pages 1 through 3 Revision 2

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.1.1 - RPS INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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 done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The NOTE to CTS 2.1.A.1.b describes action required when LHGR and MCPR limits are exceeded. This is redundant to the actions required by CTS 3.5.J and 3.5.K. Any technical changes made to these specifications will be addressed by the Justifications for Changes to Section 2.0. As such, this deletion of redundant actions is considered administrative.

A3 Note 3 to Table 4.1.A and Note 2 Table 4.1.B have been deleted since these allowances are specified in proposed SR 3.0.1.

A4 The nomenclature for frequencies have been changed as follows: "Refueling Cycle" to "18 months," "3 months" to "92 days," and "6 months" to "184 days." Since these intervals are equivalent, the proposed change is considered administrative.

A5 The Allowable Values for these Functions have been designated as NA. Current Technical Specifications leave the allowable value column blank. These changes are clarifications and as such are considered administrative.

A6 Applicability for the Turbine Stop Valve Closure and Turbine Control Valve Closure or Turbine Trip functions (ISTS Functions 8 and 9) has been changed to ≥ 30% RTP. CTS lists as MODE 1 with a note (Note 4) stating that the functions are bypassed when turbine first stage pressure is less than 154 psig. The CTS Action is to reduce power below 30% RTP (reference Notes for Table 3.1.A, Note 1.D). Section 14.5.1.5 of the FSAR, which is the plant Safety Analysis for the Bypass Valves Failure Following Turbine trip, states "Turbine first-stage pressure is used to initiate this bypass at 154 psig. The highest power level for

which these scrams remain bypassed is about 30 percent of rated power." Therefore, since 30% rated power is where bypass occurs replacing 154 psig first-stage pressure (where bypass occurs) with it is considered an administrative change.

A7 Each of the 8 MSIVs inputs its closure signal to each RPS trip system. All channels are required OPERABLE to assure a scram with the worst single failure. Therefore, the minimum channels is more appropriately specified as "8." The CTS used 4 channels based on the RPS having 4 channel trips A1, A2, B1 & B2 each of which required 4 limit switch inputs (channels) to be functional. In the ITS to better meet the definition of "channels per trip system" the total number of inputs from the valve limit switches (8) will be used. This is only a clarification in the terminology because in the CTS all inputs to the trip systems are required to be operational and in the ITS all inputs to the trip systems are required to be operational. This is administrative because it is only a clarification to better define that all of the channels per trip system are required.

A8 The existing action to "initiate insertion of OPERABLE rods . . . and fully insert all OPERABLE control rods within one hour" (see comment L1 for a change to what gets inserted) is proposed to be revised to "initiate action to insert . . . immediately." A statement on one way to reduce power is not needed. The unit must be in MODE 3 within 4 hours; thus to do so, the control rod insertion must obviously be initiated at some point. It is not necessary to state this. For MODE 5 requirements (proposed ACTION H), the existing requirement would appear to provide one hour in which control rods could be left withdrawn, even if able to be inserted. Also, if the control rod is incapable of being inserted in one hour, the existing action would appear to result in the requirement for an LER. The intent of the action is believed to be more appropriately presented in proposed Required Action H.1. With the proposed action, a significantly more conservative requirement to insert the control rod(s) and maintain them inserted is imposed. No longer would the provision to withdraw or leave withdrawn one or more control rods for up to one hour appear to exist. With this conservatism however, comes the understanding that if best efforts to insert the control rod(s) took longer than one hour, no LER would be required.

This interpretation of the intent is supported by the BWR Standard Technical Specifications, NUREG 1433. As an enhanced presentation of the existing intent, the proposed change is deemed to be administrative.

A9 These proposed changes provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry is allowed for each . . .") and "in one or more Functions" provides direction consistent with the intent of the existing Action for an inoperable RPS instrumentation channel. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative.

A10 Channel Checks have been specifically added for Functions 1.a, 1.b, 2.a, 2.b, 2.c, and 4. Although there are no channel checks required by CTS 3.1/4.1, these functions are common to CTS 3.2 functions that currently require channel checks for the same instruments. Therefore the addition of channel checks is considered administrative.

A11 The Unit 1 CTS calibration frequencies for these functions are not consistent with Units 2 and 3. These frequencies have been reflected in the proposed BFN ISTS for Unit 1 as the same as those in the proposed ISTS for Units 2 and 3. The Unit 1 Calibration frequencies for these functions will be validated prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A12 This change proposes to delete the following requirements for the RPS Functions when in MODE 5.

- The High Reactor Pressure Function will be OPERABLE with the mode switch in refuel and the reactor pressure vessel head bolted to the vessel.
- The High Drywell Pressure Function will be OPERABLE with the mode switch in refuel and primary containment integrity required.
- The Reactor Low Water Level Function will be OPERABLE with the mode switch in refuel.

The proposed change will delete the requirement for these Functions to be OPERABLE when the mode switch is in the refuel mode (even if rods are withdrawn). The High Reactor Pressure Function is not required in MODE 5 because the RCS is not pressurized and the reactor pressure vessel head is not bolted on. The High Drywell Pressure Function is not required in MODE 5 because there is not enough energy in the RCS to overpressurize the drywell and containment integrity is not required. The Reactor Low Water Level Function is not required in MODE 5 because proposed Specifications 3.9.6, "RPV Water Level," 3.9.7, "RHR-High Water Level," 3.9.8, "RHR-Low Water Level," ensure adequate cooling and retention of fission product activity. These changes are consistent

with NUREG-1433. This change is considered administrative because the requirements specified are nullified by other requirements listed in TABLE 3.1.A and associated notes. The requirement for High reactor pressure is modified by note 10 which would only require it to be operable when the reactor head is bolted to the vessel. Since part of the definition of Refuel Mode is to have at least one head bolts less than fully tensioned this requirement would not be applicable to Refueling Mode. Note 8 indicates that High Drywell Pressure is not Necessary. Note 7 (attached to Refuel header) indicates that "When the reactor is subcritical and the reactor water is less than 212°F, only the following trip functions need to be OPERABLE:

A. Mode Switch in shutdown
B. Manual scram
C. High flux IRM
D. Scram discharge volume high level
E. APRM 15 percent scram
F. Scram pilot air header low pressure (for Units 2 and 3 only)

Since the reactor will remain subcritical and less than 212°F while in the Refueling Mode this indicates that Reactor Low Water Level is not required for this mode.

Thus based on the preceding it is concluded this is an administrative change and that it is unnecessary to transfer these requirements to the ITS because they are not applicable to the Refueling Mode and their deletion makes it consistent with NUREG-1433.

A13 This proposed change will delete Note 8 from the High Drywell Pressure Function requirement when the plant is in the Startup Mode. The Note allows this Function to be inoperable when primary containment integrity is not required. Primary containment integrity, via the specifications of Section 3.6, is required in MODE 2. Therefore, the Note which allows this Function to be inoperable in MODE 2 when primary containment integrity is not required has been deleted. This change is consistent with NUREG-1433.

A14 Deleted by Revision 1.

A15 The words are clarified to provide direct indication of the intent of the current wording. Providing "at least one OPERABLE channel in the same trip system is monitoring that parameter" is intended to assure that the trip capability of that function is maintained. However, it does not provide this assurance for all logic system designs. The proposed Note will assure trip Function capability for all designs. Since this is only a clarification of the existing note it is considered an administrative change.

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These Requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 The time provided to be in MODE 2 (for the APRM function) has been reduced from 8 hours to 6 hours. This new Completion Time provides adequate time to reach MODE 2 without challenging plant systems. The time provided for the MSIV closure scram function to be in MODE 2 has been reduced from 8 hours to 6 hours. The current actions require the MSIVs to be closed in 8 hours, whereas the new ACTION (ACTION F) requires the unit to be in MODE 2 within 6 hours. In CTS, the MSIV closure function is not required to be operable in MODE 2. Therefore, this action is consistent with the current Applicability. These changes are consistent with the BWR Standard Technical Specifications, NUREG 1433, and are additional restrictions on plant operation.

M2 Current Technical Specification Action (Notes for Table 3.1.A, Item 1.D) provides no time limit for reducing power to less than 30% RTP. The proposed Action E requires this action to be completed within 4 hours and as such is more restrictive. Four hours is considered an acceptable amount of time to safely reduce power from any operating power level to 30% or less RTP. This change has been evaluated for BFN and determined to be an acceptable new restriction as it provides clarification on the amount of time allowed to achieve the required power reduction. This change is consistent with NUREG-1433.

M3 A Surveillance has been added. Proposed SR 3.3.1.1.5, which verifies that the SRM and IRM channels overlap, is required prior to withdrawing SRMs from the fully inserted position. This change will provide further assurance that the IRMs are capable of adequately monitoring neutron flux when transition from the SRMs to IRMs is being performed. The change has been evaluated for BFN and determined to be acceptable. It does not impose restrictions which would be a burden and is safe for the plant based on the current design bases. This change is consistent with the BWR Standard Technical Specifications, NUREG 1433 and is an additional restriction on plant operation.

M4 Proposed SR 3.3.1.1.9 for the IRM High Flux function has been added to require a calibration every 3 months. This addition of a new requirement constitutes a more restrictive change. This additional surveillance requirement and its associated notes are consistent with NUREG-1433.

M5 The IRM/APRM overlap is currently required to be verified during startups. The Proposed frequency will required to be performed during shutdowns (Proposed SR 3.3.1.1.6). This is an additional restriction on plant operation. Changing the requirement to verify IRM/APRM overlap from startup to shutdown is consistent with NUREG-1433. The deletion of the requirement to perform overlap verification during startup is less restrictive and is addressed by JFCL7.

M6 An RPS Channel Test Switch Function has been added to the list of functions. Appropriate Actions and Surveillance Requirements have been included for this function. The addition of this function is considered more restrictive. This function was added to allow weekly testing of the scram contactors so that other automatic scram functions' frequencies could be extended. Weekly testing of the scram contactors was credited in the analysis described in LB2 below.

M7 Proposed SR 3.3.1.1.14 verifies all logic is functioning properly by performing a LOGIC SYSTEM FUNCTIONAL TEST. These changes are consistent with the BWR Standard Technical Specifications, NUREG 1433 and are additional restrictions on plant operation.

M8 CTS Table 3.1.A, Note 17, indicates that APRM High Flux ≤15% and APRM Inoperative functions are "Not required while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MW(t)". Generally physics tests are performed as described in FSAR section 13.10 as defined in ITS section 1.1 "Definitions". One of those tests described in FSAR section 13.10 is the Shutdown Margin (SDM) Test which is addressed in ITS LCO 3.10.8 which specifies that MODE 2 functions 2.a (APRM High Flux ≤15%) and 2.e (APRM Inoperative) are required. Based on this requirement the exception allowed in Note 17 is being deleted. This is a more restrictive change and therefore further justification is not required.

For the IRM High Flux Trip Function, CTS Table 3.1.A Note 22 provides an allowance in the Refuel mode that three required IRMs per trip channel is not required if at least four IRMs (one in each core quadrant) are connected to give a noncoincidence, High Flux scram. The CTS note also states the removal of four shorting links is required to provide noncoincidence high flux scram protection from the IRMs. This provision is not included in the proposed ISTS and has been deleted. The deletion of this provision is acceptable based on the provision only being

applicable to operations during refueling with any control rod withdrawn from a core cell containing one or more fuel assemblies, at least four IRMs are still required to be OPERABLE under the provision, and the proposed ISTS allows the associated trip system to be placed in trip and activities to continue when the three required channels for a trip system are not met. This change is consistent with BWR Standard Technical Specifications, NUREG 1433 and is a more restrictive change.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 Details of the methods for performing surveillances are relocated to the Bases and procedures. The design features and system operation which dictate the methods are described in the FSAR. Additionally, changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

LA2 System design and operational details have been relocated to the Bases, procedures, and FSAR. Trip setpoints are an operational detail that are not directly related to the operability of the instrumentation. The Allowable Value is the required limitation of the parameter and this value is retained. Details relating to system design, purpose and operation (e.g., bypasses and number of inputs) are also unnecessary in the LCO and have been relocated to the Bases and procedures. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

LA3 The equipment identifier numbers from the master equipment list are relocated to the Bases. The numbers are also controlled as part of the equipment location index and on plant drawings. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the master equipment list numbers will be controlled by the licensee controlled programs.

LA4 Deleted (Deleted by Revision 2).

LA5 This is a protective trip in the Turbine and Generator. Following fast closure of the turbine stop and control valves, a reactor scram occurs if the turbine first stage pressure is greater than 154 psig. The permissive trip function is tested in conjunction with the Turbine Control Valve Fast Closure Trip function. Therefore, the permissive trip function can be relocated to the Technical Requirements Manual.

LB1 Deleted by R1.

LB2 The allowed out of service time (AOT) for placing the channel or trip system in trip is extended to 6 hours if channels in both trip systems are inoperable but trip capability is maintained (ACTION B), and 12 hours if channels in one trip system are inoperable but trip capability is maintained (ACTION A). If these new ACTIONS are not met, proposed ACTION D will direct the operator to the Table to determine the follow-on actions. The AOT for placing a channel in trip for required surveillance testing has been extended from 4 hours to 6 hours. The channel functional test frequency (STI) has been extended to once per 92 days from monthly. BFN has performed a plant specific analysis which concludes that the BFN RPS System is consistent with the GE analyses. These AOTs and STIs have been shown to maintain an acceptable risk in accordance with previously conducted reliability analyses (NEDC-30851-P-A, March 1988).

"Specific"

L1 CTS Table 3.1.A Note 7 is applicable to the Refuel Mode which is Mode 5 in ITS Table 3.3.1.1-1. Note 7 identifies six trip Functions for Units 2 and 3 and five trip Functions for Unit 1 that must be OPERABLE in the Refuel Mode. These Functions are Mode Switch in Shutdown, Manual Scram, High Flux IRM, Scram Discharge Volume High Level, APRM 15 percent Scram, and Scram Pilot Air Header Low Pressure (Units 2 and 3 only). These Functions correspond to Functions 10, 11, 1a and 1b, 7a and 7b, 2a and 2e, and 13 (Units 2 and 3 only), respectively, in ITS Table 3.3.1.1-1. These Functions, with the exception of 2a and 2e which had their applicability to Mode 5 deleted (refer to JFC L8), have all had their applicability modified to be required for Mode 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. Control rods withdrawn from a core cell containing no fuel assemblies have a negligible impact on the reactivity of the core and therefore are not required to be OPERABLE with the capability to scram. Provided all rods otherwise remain inserted, the RPS functions serve no purpose and are not required. In this condition the required shutdown margin (LCO 3.1.1) and the required one-rod-out interlock (LCO 3.9.2) ensure no event requiring RPS will occur. The actions for inoperable equipment in Mode 5 are also revised to be consistent with the interlock LCO and proposed Applicability. Since all control rods are required to

be fully inserted during fuel movement (LCO 3.9.3), the proposed applicable conditions cannot be entered while moving fuel. The only possible core alteration is control rod withdrawal which is adequately addressed by the proposed action.

L2 CTS Table 3.1.A requires Mode Switch in Shutdown, Manual Scram, High Water Level in West Scram Discharge Tank, and High Water Level in East Scram Discharge Tank trip functions in shutdown mode (modes 3 or 4) Modes 3 and 4 were not specified as applicable modes in ITS Table 3.3.1.1-1 for these functions. During normal operation in Modes 3 and 4, all control rods are fully inserted and the Reactor Mode Switch Shutdown position control rod withdrawal block (LCO 3.3.2.1) does not allow any control rod to be withdrawn. Under these conditions, the RPS function is not required to be OPERABLE and thus the deletion during normal operation is justified. Special Operations LCO 3.10.3 and LCO 3.10.4 will allow a single control rod to be withdrawn in Mode 3 or 4 by allowing the Reactor Mode Switch to be in the Refuel position. Therefore, these Mode 3 and 4 RPS functions (Mode Switch in Shutdown, Manual Scram, High Water Level in Scram Discharge Tank) are included in the Special Operations LCOs.

L3 The time to reach MODE 3 (all rods inserted) has been extended from 4 hours to 12 hours. This provides the necessary time to shutdown the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433.

L4 This change adds a note to the APRM heat balance calibration (SR 3.3.1.1.2) which states the Surveillance is not required to be met until 12 hours after Thermal Power ≥ 25% RTP. The CTS requirement is to perform a heat balance calibration once/7 days. The change is being made to be consistent with NUREG-1433 and is justified as follows. The current Technical Specifications (CTS) provides data on the methods of monitoring the core power in 3.5.I, Average Planer Linear Heat Generation Rate, and 3.5.J, Linear Heat Generation Rate, both of which must be checked daily when at ≥ 25% rated thermal power (RTP).
Section 3.5.K, Minimum Critical Power Ratio, of the CTS also indicated it must be checked daily when at ≥ 25% rated thermal power.
Section 3.5.L, APRM Setpoints, indicates that when core thermal power is ≥ 25% of rated, the ratio of FRP/CMFLPD shall be ≥ 1.0 or APRM setpoints should be adjusted accordingly. The Bases for the above CTS sections all indicate that the region of concern is ≥ 25% rated thermal power and that margin exists below that value such that the associated limits will not be violated. The change to applicability is acceptable because the heat balance calibration cannot be performed accurately at less than

25% RTP and 12 hours is a reasonable time to perform the heat balance after exceeding 25% RTP.

L5 For RPS scram functions, CTS Table 3.1.A, Note 1 requires the actions listed below for a trip function to be taken when the minimum number of OPERABLE instrument channels cannot be met by either trip system. Proposed BFN ISTS 3.3.1.1, Required Action C.1 provides a one hour Completion Time from discovery of RPS trip capability to make this declaration. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in trip capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels. This change is consistent with the BWR Standard Technical Specification, NUREG-1433. Also, the analysis (NEDC-30851P-A) upon which the RPS AOTs and STIs for BFN's ITS are based utilized the basis that for "The number of operable channels less than required by the minimum operable channels less than required by the minimum operable channels...for both trip systems, place at least one trip system in the tripped condition within one hour and take the action required by table...". Based on this it is concluded that the change is acceptable.

L6 CTS Table 3.1.A, Action, for the APRM flow bias, APRM fixed trip, APRM downscale, MSIV closure, Turbine Control Valve Fast Closure or Turbine Trip, and Turbine Stop Valve Closure trip functions allow use of CTS Action 1.A or 1.B, 1.C, or 1.D (as appropriate). Action 1.A requires Control Rods to be inserted in four hours. The proposed ITS actions for these functions are to be in Mode 2 in six hours or to reduce power to $\leq$ 30% in four hours. These actions are based on the applicability of the trip functions. CTS Action 1.A option has been deleted to be in agreement with NUREG-1433. This is less restrictive but still acceptable because it was one of two acceptable actions. The preferred action has been retained. Since the preferred option was already available, this will not have an impact on safety. The option that was previously available (1.A) will still be available to the operator should they choose to exercise it.

L7 The IRM/APRM overlap is required by CTS Table 4.1.B, Note 6, to be verified during controlled startup. This requirement is being deleted and a requirement to verify overlap when entering Mode 2 from Mode 1 (during shutdown) is being included in ITS 3.3.1.1.6. The change from doing the overlap verification during shutdown (see JFC M5) rather than startup is acceptable because during startup the system design will prevent increases by initiating a rod block if adequate overlap is not maintained. This change is consistent with the BWR Standard Technical Specifications, NUREG-1433.

L8 The CTS requirements for the APRM High FLux (Setdown) - (≤ 15% rated power) Function and APRM Inoperative Function to be Operable in the Refuel Mode (Mode 5) has not been retained in the Improved Technical Specifications. These Functions are not required because the IRMs are capable of performing the required trip function and the SRMs are capable of indicating impending criticality and producing a Rod Block Signal. Since only one rod is permitted in the Refuel Mode and there is adequate margin to assure the core remains noncritical with one rod out and withdrawal of a second rod is prevented by refuelling interlocks and administrative controls the IRMs and SRMs provide adequate instrumentation without these APRM Functions to preclude violation of a safety limit. This change is consistent with NUREG-1433.

L9 The calibration frequency for Local Power Range Monitors (LPRMs) is being changed from 1000 Effective Full Power Hours (EFPH) to 1000 Megawatt-Days per Ton (MWD/T) average core exposure to match NUREG-1433. This change constitutes an approximate 11 percent increase in the interval between calibrations. BWR power operation relies upon readings from fixed incore neutron detectors known as Local Power Range Monitors (LPRM). LPRMs are small fission chambers with an approximately linear response to the local neutron flux and, thus, local thermal power. The current Surveillance Requirement to calibrate the LPRMs every 1000 EFPH employs a second set of movable detectors known as the Traversing Incore Probe (TIP) System. The required LPRM calibration relates the power distribution, measured by the TIP System, to the then existing LPRM readings. When the LPRMs are normalized to one another, to the TIP readings, and to a plant heat balance calculation, these LPRMs allow determination of the local power for each (approximately) six inch fueled region of the core (node).

Outputs from the calibrated LPRMs are used in the Reactor Protection System (Average Power Range Monitor) and the Rod Block Monitor, as well as for daily surveillance of Power Distribution Limits (reactor thermal limits monitoring). Accuracy requirements on the power distribution are defined by GESTAR-II, (NEDE-24011-P-A-10, Section 4.3.1.1.1) and GE Fuel Bundle Designs, NEDE-3112P, which are part of the fuel licensing basis. In particular, Table 3-3 of NEDE 3112P requires calculated nodal powers to have a root mean square (rms) uncertainty of no more than 8.7% for reload cores. The attending Table 3-3 comment states that this uncertainty also applies to the power distribution as determined by the LPRM System between TIP sets and must also meet the 8.7% rms uncertainty.

Advances in process computer monitoring include the development of new mathematical techniques and algorithms combining reactor physics theory with online core data, (e.g., LPRM readings). One such methodology presently employs an adaptive learning algorithm using online as well as

historical core data inputs to improve power calculations within the reactor physics model by effectively modifying the neutron leakage terms (adaptive coefficients) to force the calculated power distribution to match the measured power distribution as determined by the TIP System. An adequate number of LPRM calibrations are performed upon startup and initial operation to establish a base set of adaptive coefficients. Subsequent calculations use the adaptive coefficients and LPRM readings during monitoring between LPRM calibrations. The set of adaptive coefficients is updated at each LPRM calibration throughout the cycle.

Corrections made within the monitoring process account for decay of LPRM sensitivity due to depletion of the fissile coating within each LPRM. The 1000 EFPH calibration interval was based upon the older monitoring methodology and older LPRM designs in use at the time. Calibration current and exposure data demonstrate a significant reduction in the uncertainty associated with LPRM sensitivity as a function of exposure. This reduced uncertainty, combined with improved monitoring methods, allows lengthening of the LPRM calibration interval.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.1.2 - SRM INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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 done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as more Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These requirements are based on the Standard Technical Specifications for BWR/4, NUREG 1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 The MODE 2 SRM requirements have been modified to require three SRMs instead of the current two SRMs. This is an additional restriction on plant operation and is consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.

M2 Existing Specifications 3.3.B.4 and 4.3.B.4 require SRMs to be operable whenever control rods are withdrawn for startup or refueling. Proposed LCO 3.3.1.2 (Table 3.3.1.2-1) will require SRMs to be operable at all times in MODE 2 prior to and during control rod withdrawal until flux level is sufficient to maintain the Intermediate Range Monitor (IRM) on Range 3 or above. This more restrictive change is consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.

M3 Additional requirements have been added to ensure two SRMs are OPERABLE during MODE 3 and MODE 4. This ensures flux monitoring is available while shutdown. Appropriate ACTIONS (ACTION D) and Surveillance Requirements (SRs 3.3.1.2.3, 3.3.1.2.4, 3.3.1.2.6, and 3.3.1.2.7) have also been added. These new requirements are additional restrictions on plant operation and are consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.

M4 Additional Surveillance Requirements have been added to ensure SRM OPERABILITY. Proposed SR 3.3.1.2.1 requires a CHANNEL CHECK to be performed every 12 hours during MODE 2. Proposed SR 3.3.1.2.6 requires a CHANNEL FUNCTIONAL TEST to be performed and signal to noise ratio be determined every 31 days during MODE 2. Proposed SR 3.3.1.2.7 requires a CHANNEL CALIBRATION to be performed every 92 days during MODE 2 and MODE 5. In addition, a requirement to determine signal to noise ratio has been added to existing CTS 3.3.B.4 (Proposed SR 3.3.1.2.4). These new Surveillance Requirements are additional restrictions on plant operation. The proposed changes are consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.

M5 Proposed LCO 3.3.1.2 (Table 3.3.1.2-1) will require that channel functional tests be performed every 7 days when in MODE 5 instead of prior to CORE ALTERATIONS as is currently required in CTS 4.10.B.1. SR 3.3.1.2.5 will also add a requirement to determine signal to noise ratios once per 7 days. Additionally, proposed LCO 3.3.1.2 (Table 3.3.1.2-1) will require that Channel Checks (SR 3.3.1.2.1) be performed every 12 hours when in MODE 5 instead of prior to CORE ALTERATIONS as is currently required in CTS 4.10.B.1. Proposed SRs 3.3.1.2.1 and 3.3.1.2.5) are more restrictive than existing specifications. The proposed changes are consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.

M6 CTS 3.10.B.1 requires verification during CORE ALTERATIONS that an operable SRM detector is located in the following locations: the fueled region; the core quadrant where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region; and the core quadrant adjacent to where CORE ALTERATIONS are being performed, when the associated SRM is included in the fueled region. Proposed SR 3.3.1.2.2 has the same requirements; however, SR 3.3.1.2.2 will require periodic verification at least once per 12 hours. This is more restrictive on plant operation. The proposed change is consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.

M7 Proposed ACTION E is provided to ensure proper actions are taken during MODE 5 operations, including CORE ALTERATIONS. The proposed ACTIONS require suspension of CORE ALTERATIONS, except for rod insertion, and require control rods in core cells containing fuel assemblies to be inserted. This is an additional restriction on plant operation that is consistent with the BWR/4 Standard Technical Specifications, NUREG 1433.

M8 CTS 3.3.B.4 does not identify Required Actions if SRM Operability requirements in Mode 2 are not satisfied. Proposed LCO 3.3.1.2 will identify the Required Actions and associated Completion Times if SRM Operability requirements in Mode 2 are not satisfied. Proposed Condition A will allow 4 hours to restore the 3 required SRM channels to Operable as long as at least one SRM is always Operable. Proposed Condition B will require suspension of all control rod withdrawal if there are no Operable SRMs; and, in accordance with Condition A, will allow 4 hours to make the required 3 SRM channels Operable. Proposed Condition C will require that the reactor be in Mode 3 within 12 hours if Required Actions and Completion Times for Condition A or B are not satisfied. Proposed Conditions A, B, and C are more restrictive than the existing specifications since proposed Conditions A, B and C allow up to 16 hours (4 hours for Conditions A and B and 12 hours for Condition C) before the reactor must be in Mode 3 when SRM Operability requirements are not satisfied versus the CTS requirement of not withdrawing any control rods. These more restrictive Required Actions are consistent with BWR/4 Standard Technical Specifications, NUREG-1433.

M9 CTS 4.10.B.2 currently requires "Once per 12 hours during CORE ALTERATIONS, verify that the associated SRM (FLC) is reading ≥ 3 cps with a signal-to-noise ratio ≥ 3:1". A change to the SR as it is being transferred to the ITS (SR 3.3.1.2.4) is also to verify the ≥ 3 cps with a signal-to-noise ratio ≥ 3:1 every 24 hours. This additional requirement will provide added assurance during times when no reactivity changes are being made that the SRMs are indicating count rate indicative of flux levels within the core. This is a more restrictive change that is being added to be consistent with the BWR Standard Technical Specifications, NUREG-1433.

<u>TECHNICAL CHANGE - LESS RESTRICTIVE</u>

"Generic"

LA1 System design and operational details have been relocated to the Bases and procedures. Details of the requirements of SRM OPERABILITY have been relocated to the Bases and procedures. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

"Specific"

L1 If a spiral offload or reload pattern is used, the proposed specifications will allow a reduction in the number of SRM channels required to be operable during refueling. Specifically, existing Specification 3.10.B.1 requires two SRMs during Core Alterations. Proposed Specification 3.3.1.2 (Table 3.3.1.2-1 footnote (b)) reduces the number of SRM channels required to be operable from 2 to 1 "during spiral offload or reload when the fueled region includes only that SRM detector." A reduction in the number of required operable SRM channels is acceptable when using a spiral pattern for loading or offloading fuel because the use of a spiral pattern provides assurance that the operable SRM is in the optimum position for monitoring changes in neutron flux levels resulting from the Core Alteration. Additionally, existing Specification 3.10.B.2 permits the SRM count rate to fall below the specified minimum level if all control rods in cells that contain fuel are fully inserted and electrically disarmed. These changes are consistent with BWR Standard Technical Specifications, NUREG-1433.

L2 ITS Table 3.3.1.2-1, Note (a), which modifies the applicability of SRMs in Mode 2 to only being required when IRMs are on Range 2 or below, is not specified in the CTS. It is acceptable to incorporate this Note because when the IRMs are on Range 3 or above, they provide adequate monitoring of flux levels. The SRMs are intended to provide flux data at low flux levels to monitor the approach to criticality. After the flux levels are high enough to allow SRM/IRM overlap, the SRMs are normally withdrawn. BFNs FSAR states, "During reactor startup SRM detectors may be withdrawn after the neutron flux has sufficient indication on the IRMs." Thus, while the CTS did not specifically specify the applicability, the adoption of Note (a) to Table 3.3.1.2-1 is acceptable because the IRMs provide adequate flux monitoring capability after they are on Range 3 and the SRMs are withdrawn at the higher flux levels. This change is consistent with BWR Standard Technical Specifications, NUREG-1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.2.1 - CONTROL ROD BLOCK INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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 done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 Last sentence of Note 5 deleted since it is only an informational reference.

A3 Currently the only requirement specified in CTS Tables 3.2.C and 4.2.C for the rod block logic function is from Table 4.2.C under functional test which references 4.2.C Note 16. Note 16 stated "Performed during operating cycle. Portions of the logic is checked more frequently during functional test of the functions that produce a rod block." There are 19 functions that initiate rod blocks listed in Tables 3.2.C and 4.2.C. Each of functions has a functional test that verifies the function produces a rod block which also verifies the rod block logic for the required functions. The longest functional test frequency for these functions is once per operating cycle which is equivalent to the frequency specified in Note 16 for the logic functional. Most of the functions are more frequent than once per cycle. Each of these functions verify rod block logic for its particular function. For the functions listed in ITS Table 3.3.2.1-1 each has a SR to perform functional test which will verify rod block logic. The longest frequency for the ITS functional test is 18 months (equivalent to once per operating cycle). Thus, the rod block logic will be accomplished by functional test in the ITS similar to the CTS. DOC R1 provides justification for many of the functions in CTS Tables 3.2.C and 4.2.C not being transferred to ITS Table 3.3.2.1-1. Since the omissions were justified as not being required in the ITS, all of the necessary rod block logic functional tests will still be performed by the functional tests included in the ITS requirements. Thus, it is concluded that rod block logic functional test requirements in the CTS are appropriately addressed in the ITS for all functions retained in the ITS.

A4 For Specification 3.3.2.1, these Current Technical Specification notes are not used. These Notes will be addressed as appropriate in other applicable Specifications.

A5 Deleted (A5 Deleted by R1).

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type DOCs) contain requirements that are more restrictive than the Current Technical Specifications. These requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 The Reactor Mode Switch Shutdown Position Function has been added. MODE 3 and 4 requirements ensure that all rods remain inserted when the mode switch is in shutdown. In MODE 5 with the mode switch in shutdown, the control rod withdrawal blocks are assumed in the safety analysis to prevent criticality. Therefore, they must be OPERABLE to fulfill the safety analysis. An applicable ACTION (ACTION E) and a Surveillance Requirement (SR 3.3.2.1.6) have also been added.

M2 An additional Surveillance has been added for the RWM. Proposed SR 3.3.2.1.5 ensures the automatic enabling point of the RWM is calibrated properly. This is an additional restriction on plant operation.

M3 Existing specifications require that the Rod Block Monitor must be OPERABLE: "During operation with CMFCP or CMFLPD equal to or greater than 0.95" (3.3.B.5); "RBM need not be OPERABLE in the 'Startup' position" (Table 3.2.C, Note 7.b); and, "The RBM channels are bypassed when reactor power is $\leq$ 30% or when a peripheral (edge) control rod is selected" (Table 3.2.C, Note 7.a). Proposed Specification 3.3.2.1, Control Rod Block Instrumentation, applicability is during operation with thermal power $\geq$ 29% and < 90% and MCPR < 1.75 or during operation $\geq$ 90% RTP and MCPR < 1.44. Thus, the RTP limit has been reduced from 30% to 29% with additional restrictions on MCPR. The peripheral (edge) control rod selected bypass is a system design detail that has been incorporated into the ITS Bases (see DOC LA1). The allowance that the RBM need not be OPERABLE in the Startup position has been eliminated. Changes from the BWR Standard Technical Specifications, NUREG-1433, are

because ARTS/MELLA have not been implemented at this time although the analyses are complete to support the implementation. Based on the above, it is concluded that these proposed changes are more restrictive.

M4 The frequency for proposed SR 3.3.2.1.4 has been changed from six months to 92 days. This change is necessary to be consistent with other calibration frequencies for the related APRM instrumentation which has a 3 month (92 day) calibration frequency specified in a plant specific setpoint calculation. This is a more restrictive change.

M5 CTS Table 3.2.C, Note 5, states "During repair or calibration of equipment, not more than one ... RBM channel ... may be bypassed. Bypassed channels are not counted as operable channels to meet the minimum operable channel requirements." Note 2 to the Surveillance Requirements (less restrictive aspects addressed by LA1) for the ITS only allows six hours for calibration and does not address repair. Since no timeframe is specified for bypassing the channel versus six hours in the ITS and there is no allowance in the ITS for maintenance, the ITS is more restrictive than the CTS.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 System design and operational details have been relocated to the Bases and procedures. Trip setpoints are an operational detail that is not directly related to the operability of the instrumentation. The Allowable Value is the required limitation of the parameter and this value is retained. Details relating to system design and operation (e.g., description of inoperable trip) are also unnecessary in the LCO and have been relocated to the Bases and procedures. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

LA2 Details of the purpose and methods of performing surveillances are relocated to the Bases and procedures. The design features and system operation which dictate the methods are described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

LB1 The Channel Functional Test Frequency has been changed from 1 month to 92 days. Browns Ferry has performed a plant specific evaluation of the RBM system that concluded the BFN RBM system is consistent with the GE analysis performed in NEDC-30851-P-A, October 1988. This plant specific analysis justifies the extension of the functional test from monthly to quarterly.

"Specific"

L1 All proposed Surveillances for LCO 3.3.2.1, Control Rod Block Instrumentation, will include a Note stating "When an RBM is placed in an inoperable status solely for performance of a required Surveillance, entry in the associated Conditions and Required Actions may be delayed up to 6 hours provided the associated Function maintains rod block capability." This change is acceptable because the Note applies only when the rod block function is maintained by the redundant RBM channel, the six hour period is shorter than the allowable outage time (AOT) of 24 hours, and tests are conducted infrequently. This change is consistent with BWR Standard Technical Specifications, NUREG-1433.

L2 Existing Specifications 4.3.B.3.b.1 & 2 require a Channel Functional Test of the Rod Worth Minimizer (RWM) prior to startup and prior to shutdown. Proposed Specification 3.3.2.1 will require a Channel Functional Test of the RWM every 92 days in Mode 2 and every 92 days in Mode 1 when Thermal Power is $\leq$ 10%. Proposed SR 3.3.2.1.2 will be modified by a Note stating that the Channel Functional Test is not required during a startup until 1 hour after any control rod is withdrawn at $\leq$ 10% RTP in Mode 2. Because the Channel Functional Test is not required until after startup in the proposed ITS, Note 4 to CTS Section 1.0.M, which allows placing the reactor mode switch in startup for RWM testing, can be deleted. CTS 4.3.B.3.b.1.b & c require the test within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical. Proposed SR 3.3.2.1.3 will be modified by a Note stating that the Channel Functional Test is not required during a shutdown until 1 hour after Thermal Power $\leq$ 10% in Mode 1. CTS 4.3.B.3.b.2.b requires verification of proper annunciation of the selection error of at least one out-of-sequence control rod within 8 hours prior to RWM automatic initiation when reducing thermal power. However, CTS 4.3.B.3.b.2.c does allow the rod block function to be verified within one hour after RWM automatic initiation. The addition of these Notes makes the proposed requirement for a Channel Functional Test less restrictive because the Surveillance Test is not required until 1 hour after the RWM is required to be Operable. These changes are acceptable for the following reasons: a) the Rod Worth Minimizer does not monitor core thermal conditions but simply enforces preprogrammed rod patterns as a backup and is intended to prevent reactor operator error in selecting or positioning control rods; b)

reliability analysis documented in NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988, determined that the failure frequency curve for this instrumentation is relatively flat in the range of 30 to 124 days and starts a gradual increase after 124 days which means that more frequent testing is unlikely to identify problems; and c) it is overly conservative to assume that the RWM is not operable when a surveillance is not performed because of its demonstrated reliability as demonstrated by successful completion of most Channel Functional Tests. This change is consistent with BWR/4 Standard Technical Specifications, NUREG-1433.

L3 CTS inoperability requirements for the RBM channels are detailed in Table 3.2.C, Note 7.c and 7.d, and LCO 3.3.B.5 which state:

7.c Two RBM channels are provided and only one of these may be bypassed from the console. If the inoperable channel cannot be restored within 24 hours, the inoperable channel shall be placed in the tripped condition within one hour.

7.d With both RBM channels inoperable, place at least one inoperable rod block monitor channel in the tripped condition within one hour.

3.3.B.5 During operation with CMFCP or CMFLPD equal to or greater than 0.95, either:

a. Both RBM channels shall be OPERABLE:
or
b. Control rod withdrawal shall be blocked.

These requirements are translated into the ITS LCO 3.3.2.1 "The control rod block instrumentation for each Function in Table 3.3.2.1-1 shall be OPERABLE" and these actions:

CONDITION		REQUIRED ACTION		COMPLETION TIME
A.	One rod block monitor (RBM) channel inoperable	A.1	Restore RBM channel to OPERABLE status.	24 hours
B.	Required Action and associated Completion Time of Condition A not met. OR Two RBM channels inoperable.	B.1	Place one RBM channel in trip.	1 hour

The requirements of CTS Table 3.2.C, Note 7.c and 7.d , are all included in ITS LCO 3.3.2.1, Actions A.1 and B.1 with the exception of "and only one of these may be bypassed from the console" which was relocated and addressed by DOC LA2.

The requirements of CTS LCO 3.3.B.5 are not completely encompassed by the ITS actions. CTS LCO 3.3.B.5 does not have any time limits listed for blocking control rod withdrawal if CMFCP or CMFLPD are equal to or greater than 0.95 and both RBM channels are not OPERABLE which implies an immediate action. However, the CTS Bases for LCO 3.3.B.5 states "Two RBM channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Automatic rod withdrawal blocks from one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. The specified restrictions with one channel out of service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists." This is consistent with the Bases for CTS Table 3.2.C which states "When the RBM is required, the minimum instrument channel requirements apply. These requirements assure sufficient instrumentation to assure the single failure criteria is met. The minimum instrument channel requirements for the RBM may be reduced by one for maintenance, testing, or calibration. This does not significantly increase the risk of an inadvertent control rod withdrawal, as the other channel is available, and the RBM is a backup system to the written sequence for withdrawal of control rods." The similarity of CTS Bases for LCO 3.3.B.5 to the Table 3.3.C Bases implies that the Table 3.2.C allowed out of service times were intended to be applied to LCO 3.3.B.5. Thus, it is concluded that although not specifically called out the time limits in Table 3.2.C, Note 7.c and 7.d were intended to be applicable to LCO 3.3.B.5. The

applicability of when the RBM must be OPERABLE specified in ITS exceeds the requirement that "During operation with CMFCP or CMFLPD equal to or greater than 0.95" and is addressed by DOC M3.

L4 The proposed change eliminates Specification 4.3.B.5 which requires a Functional Test of the Rod Block Monitor (RBM) "prior to withdrawal of the designated rod(s)" during operation with CMFCP or CMFLPD equal to or greater than 0.95 and relies completely upon the Functional Test which is required every 92 days. The proposed change is acceptable because two independent RBM channels will be OPERABLE during any rod withdrawal except for short and infrequent periods when one channel is inoperable, and deletion of this requirement allows taking credit for routine periodic tests in place of performing unscheduled testing whenever the potential exists that the RBM may be required to function. The Frequency of 92 days for the Channel Functional Test is based upon the reliability analysis in NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988. This reliability study found that the failure frequency curve for this type of instrumentation is relatively flat in the range of 30 to 124 days and starts a gradual increase after 124 days. Based on this finding, performing this testing more frequently than every 92 days does not significantly increase the probability of detecting a random failure of the RBM. This change is consistent with BWR Standard Technical Specifications, NUREG-1433.

RELOCATED SPECIFICATIONS

R1 The APRM, SRM, IRM and scram discharge volume control rod blocks function to prevent a control rod withdrawal error at power transient. However, no design basis accident or transient takes credit for rod block signals initiated by this instrumentation. Reactor recirculation flow increases cause an increase in neutron flux which is monitored by the neutron monitoring system which also has the capability for providing a rod block.

This instrumentation is also not credited for rod block signal initiation following a design basis accident or transient. Further, the evaluation summarized in NEDO-31466 determined the loss of the this instrumentation to be a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for these Functions are being relocated to the Technical Requirements Manual. Any change to this instrument function will be controlled by the provisions of 10CFR50.59.

R2 The current Technical Specifications require instrument channel checks. This test is performed by a comparison of redundant channels as a simple check of instrument performance. NUREG-1433 has no equivalent check for the RBM so performance of the daily "Instrument Check" will be relocated to plant procedures and controlled in accordance with 10 CFR 50.59.

Browns Ferry Units 1, 2, & 3 were participants in NEDC-30851P-A, Supplement 1, Licensing Topical Report, Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation. The rational for not including the Instrument Channel Check in the ITS was based on this analysis which included "an acceptable format for proposed TS changes" based on the Safety Evaluation Report (SER) cover letter. The proposed format for "CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS" has the Rod Block Monitor Channel Check column marked N.A. (Not Applicable). Although, the requirement for Instrument Channel Checks is not being transferred to the ITS from the CTS, the Technical Requirements Manual will maintain the daily Channel Check requirement.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.3.1
POST-ACCIDENT MONITORING (PAM) INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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 done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 Three Notes have been provided which modify the Actions of the PAM Specification. Note 1 states the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when PAM instrumentation is inoperable. This allowance is provided due to the passive function of the instruments, the operator's ability to diagnose an accident using alternate instruments and methods and the low probability of an event requiring the use of these instruments. Adding Note 1 is considered an administrative change because existing BFN Technical Specifications do not have a requirement that prohibits entry into a MODE or condition when an LCO required by that MODE or condition is not satisfied. Therefore, existing Technical Specifications already allow the actions permitted by Note 1. Notes 2 and 3 provide more explicit instructions for proper application of the Actions for Technical Specifications compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry is allowed for each . . .") and "one or more automatic Functions" provides direction consistent with the intent of the existing Action for an inoperable isolation instrumentation channel. Since Notes 2 and 3 only provide more explicit direction of the current interpretation of the existing Technical Specifications, this change is considered administrative. In addition, Note 3 is only added in conjunction with the addition of Function 6 as described in Justification for Change M1 below.

A3 These Notes are common to several instrumentation tables and will be discussed, as appropriate, with the instrumentation they correspond to. Notes 24, 25, 26, and 30 (32 for Unit 3) are the only Notes that apply to PAM instrumentation.

A4 The frequencies have been changed from once per month to once per 31 days, once per 6 months to once per 184 days and once/cycle to once/18 months. These frequencies are considered equivalent. BFN's current operating cycles are approximately 18 months. Therefore, this change is considered administrative.

A5 CTS Table 3.2.F lists one as the minimum number of operable channels for the high range containment radiation monitor. However, based on the accompanying action notes (7 & 8), the intent is for two to be operable. The CTS is written such that a Special Report is to be submitted (Note 7) if one of the two monitors is inoperable for more than 7 days. Note 8 requires the preplanned alternate method of monitoring to be initiated within 72 hours when both monitors are inoperable. The ISTS format is such that both channels are listed as required operable. The corresponding ACTIONS require a Special Report be submitted within 14 days of one channel being inoperable for more than 30 days (ACTION B) and two channels being inoperable for more than 7 days (ACTION G). The requirement for use of alternate methods of monitoring has been moved to the Bases and procedures (Refer to justification LA2 for this Specification). The less restrictive aspects of this change are justified in Justification L5 and L6 below.

A6 CTS Table 4.2.F specifies a 6 month calibration frequency for Item 15 (Containment Atmosphere Monitor). However, CTS 4.7.H.1 requires the CAM - Hydrogen Analyzer to be calibrated quarterly. Since the CAM - Hydrogen Analyzer requirements of CTS 4.7.H.1 must be met, the 6 month calibration frequency has no meaning and has been marked as 92 days (quarterly) for consistency. The same calibration test, with a frequency of 92 days, is used to satisfy both of these requirements.

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These Requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 Requirements for additional PAM Functions and Channels are incorporated. These are included in accordance with NUREG 1433 guidelines to include all Type A and Category 1 PAMs.

The development of the BFN plant specific PAM functions is documented by the numerous Background References in Design Criteria (DC) BFN-50-7307, Browns Ferry Nuclear Plant Post-Accident Monitoring. Among those references are the following SERs:

Letter from Suzanne Black, Assistant Director for Projects, TVA Projects Division, Office of Special Projects to S. A. White, Manager of Nuclear Power, Tennessee Valley Authority dated June 23, 1988.

Letter from Suzanne Black, Assistant Director for Projects, TVA Projects Division, Office of Nuclear Reactor Regulation to Oliver D. Kingsley, Senior Vice President of Nuclear Power, Tennessee Valley Authority dated February 8, 1990.

And

Letter from Frederick J. Hebdon, Director Project Directorate II-4, Division of Reactor Projects, Office of Nuclear Reactor Regulation to Dr. Mark O. Medford, Vice President of Technical Support, Tennessee Valley Authority dated May 27, 1993.

The addition of the functions not included in the CTS in the ITS is a more restrictive change made to be consistent with NUREG-1433.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 Details of the system OPERABILITY requirements, description of the instruments, and methods to perform the Surveillances are relocated to the Bases, procedures, and the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

LA2 The statement in Note (8)of CTS Table 3.2.F relative to the initiation of alternate monitoring methods when the Primary Containment Area Radiation monitors are inoperable has not been included in proposed ACTIONS of ITS Section 3.3.3.1. However, the requirement for initiation of alternate monitoring is implied by proposed ITS 5.6.6, "PAM Report," which is required to be initiated by proposed ACTIONS B.1 and G.1 of

LCO 3.3.3.1 when inoperable channels have not been returned to operable status within the allowed Completion Times. Proposed ITS 5.6.6 requires a report to be submitted which outlines the alternate method of monitoring. The proposed ITS Bases for ACTIONS B.1 and G.1 of LCO 3.3.3.1 provide additional discussion of proposed Specification 5.6.6 and the initiation of alternate monitoring methods.

"Specific"

L1 The Frequency for the CHANNEL CHECK and CHANNEL CALIBRATION is being changed to every 31 days and every 18 months, respectively. These instruments are highly reliable, and they are providing indication only. No automatic actions are performed by this instrumentation. The sensors are also similar to others that are calibrated every 18 months. This Frequency is also consistent with the BWR Standard Technical Specifications, NUREG 1433.

L2 The Required Action for one channel inoperable in one or more Functions for more than 30 days is revised from requiring a shutdown to requiring a Special Report in accordance with the Administrative Controls Section of the Technical Specifications. This report discusses the alternate method of monitoring and the results of the root cause evaluation of the inoperability and identifies proposed restorative actions. Due to the passive function of this instrumentation and the operator's ability to respond to an accident utilizing alternate instruments and methods for monitoring, it is not appropriate to impose stringent shutdown requirements for out of service instrumentation.

L3 PAMs are provided to assist in the diagnosis and preplanned actions required to mitigate design basis accidents which are assumed to occur in MODES 1 and 2. The probability of an event in MODES 3, 4 or 5 that would require PAM instrumentation is sufficiently low that the PAMs are not required in these MODES. Therefore, the Action to be in MODE 4 within 24 hours if the functions are not restored to OPERABLE status within 6 hours of the Completion Time has been changed to be in MODE 3 within 12 hours.

L4 CTS 3.7.H.1 states the hydrogen analyzers are required to be operable whenever the reactor is not in Cold Shutdown (equivalent to requiring them to be operable in MODES 1, 2 & 3 in ISTS), while the CTS actions (3.7.H.2 & 3) for not restoring hydrogen analyzers is to be in Hot Shutdown (equivalent to MODE 3 in ISTS). The required action does not place the reactor in a condition that does not require operability of the hydrogen analyzers. The proposed applicability for 3.3.3.1 is MODES 1 and 2. Proposed Action D allows 72 hours to restore one required hydrogen monitor channel when both are inoperable, whereas current

technical specifications allow no time. However, the time allowed to reach Hot Shutdown (MODE 3) has been reduced to 12 hours. Overall, the proposed change is less restrictive. The changes to the allowed outage times are considered acceptable based on Operable diverse instrument channels, the passive nature of the instruments (no required automatic action) and the low probability of an event requiring PAM instrumentation during the intervals.

L5 The Action for a single inoperable Primary Containment Area Radiation channel has been revised. Thirty days are proposed to allow for restoration of the inoperable channel per proposed Condition A. The change from 7 days for restoration of the inoperable channel to 30 days is acceptable based on the availability of the remaining Operable Primary Containment Area Radiation channel, the passive nature of the instrument (no required automatic action) and the low probability of an event requiring the PAM instrumentation during the interval.

L6 The Actions have been changed for two Primary Containment Area Radiation channels inoperable. Seven days are proposed to be allowed for restoration of one channel prior to initiating the alternate method of monitoring, instead of the existing requirement for initiation of the alternate method of monitoring within 72 hours and restoration of two channels to Operable status. The Completion Time of 7 days for restoration of one channel or initiation of the alternate method of monitoring is considered acceptable based on the relatively low probability of an event requiring PAM instrumentation, the passive function of the instruments, and the availability of alternate means to obtain the information.

L7 The current submittal requirement for a Special Report is within 7 days of exceeding the allowed outage time for an inoperable Primary Containment Area Radiation channel(s). The proposed change will increase the submittal time from within 7 days of exceeding the time limit to within 14 days of exceeding the time limit. The proposed change is a change to the allowed time for submittal of a Special Report per Section 6.9.2 of CTS (equivalent to 5.6.6 of ITS) if required indication instruments are inoperable for periods exceeding specified time limits. This change in timeframe for submittal of a report will have no impact on the ability to mitigate an accident or an analyzed event. The function performed by the indication instruments will still be performed as is evidenced by the statement in the ITS that the Special Report shall contain "the preplanned alternate method of monitoring." The change will not affect any assumptions contained in the safety analysis. Based on the preceding it is concluded that the change is acceptable. This change is made to be consistent with NUREG-1433.

L8 The CTS Surveillance Requirement 4.7.H.2 that "Each hydrogen analyzer system shall be demonstrated OPERABLE by performing a CHANNEL FUNCTIONAL TEST monthly" is not incorporated into the ITS. This has the effect of changing the CHANNEL FUNCTIONAL TEST frequency from monthly to every 92 days (quarterly) because the CHANNEL CALIBRATION by definition includes a CHANNEL FUNCTIONAL TEST. In addition to the CHANNEL CALIBRATION there is a CHANNEL CHECK every 31 days (monthly) which would identify any significant operability problems with the analyzer because the indicated values compared between the two analyzer systems would not be similar if one were not OPERABLE. Injection of a signal into a analyzer and verification of a signal would provide minimal information beyond that of the CHANNEL CHECK. Based on the above it is concluded that the CHANNEL CHECK every 31 days and CHANNEL CALIBRATION every 92 days is adequate to verify that the operability of the analyzers and incorporation of the CHANNEL FUNCTIONAL TEST into the CHANNEL CALIBRATION frequency is acceptable. Also the change is consistent with the NUREG-1433 format because it did not include any CHANNEL FUNCTIONAL TESTS.

RELOCATED SPECIFICATIONS

R1 Current Technical Specification 3.2/4.2.H, Flood Protection, and 3.2/4.2.I, Meteorological Monitoring Instrumentation, requirements are being relocated to the Technical Requirements Manual. Any change to these instrument functions will be controlled by the provisions of 10CFR50.59.

R2 The Suppression Chamber Air Temperature, Control Rod Position, Neutron Monitoring, Drywell Pressure alarm at 35 psig, Drywell Temperature and Pressure and Timer alarm, CAD Tank Level, Drywell to Suppression Chamber Differential Pressure, Relief Valve Tailpipe Temperature or Position Indication, and Wide Range Gaseous Effluent Radiation Monitor are not credited as Category 1 or Type A variables. Further, the loss of these instruments is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for these Functions are being relocated to the Technical Requirements Manual. Any change to these instrument functions will be controlled by the provisions of 10CFR50.59.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.4.1 - EOC-RPT INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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 done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 These proposed changes provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry is allowed for each . . .") provides direction consistent with the intent of the existing Action for an inoperable instrumentation channel. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative.

A3 The Frequency of "once/operating cycle" has been changed to "18 months." This change is administrative since 18 months is an operating cycle.

A4 These Notes are common to several instrumentation tables and will be discussed, as appropriate, with the instrumentation they correspond to. Note 17 is the only Note that applies to EOC-RPT instrumentation.

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type DOCs) contain requirements that are more restrictive than the Current Technical Specifications. These requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 New Surveillance Requirements are proposed to be added. SR 3.3.4.1.3 requires a CHANNEL CALIBRATION and a verification that the Allowable Values are not exceeded. SR 3.3.4.1.2 verifies the bypass capability is properly functioning. These Surveillance Requirements (SRs) were proposed to be added to the EOC-RPT to be consistent with the Surveillance Requirements for the Reactor Protection System (RPS) for the same initiating signals because they are from the same devices (EOC)-RPT signals derived from the RPS signals). Thus while the proposed SRs are new for the EOC-RPT they were existing for the RPS and thus scope and frequency are adequate. Since these are new requirements for EOC-RPT they are considered more restrictive. These changes are consistent with the BWR Standard Technical Specifications, NUREG-1433.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 System design and operational details have been relocated to the Bases. Trip setpoints are an operational detail not directly related to the operability of the instrumentation. The Allowable Value is the required limitation of the parameter and this value is retained. Details relating to system design and operation (e.g., bypasses, when not to place in trip) are also unnecessary in the LCO and have been relocated to the Bases. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR will be controlled by the provisions of 10 CFR 50.59.

LB1 The Channel Functional Test Frequency has been changed from 1 month to 92 days, consistent with that provided in GENE-770-06-1. Browns Ferry has performed a plant specific evaluation and concluded that this instrumentation is consistent with the GE analysis performed in GENE-770-06-1. This plant specific analysis justifies the extension of functional tests for EOC-RPT instrumentation from monthly to quarterly.

"Specific"

L1 A Surveillance Requirement Note has been added to allow testing for up to 6 hours prior to requiring entry into associated Conditions and Required Actions provided the associated Function maintains EOC-RPT trip capability. Current Technical Specifications Table 3.2, Note 17, allows two hours prior to requiring that the system (channel in this case) be declared inoperable and actions taken.

L2 An option is provided for one or more inoperable channel(s) to place all inoperable channels in the tripped condition (Required Action A.2). This conservatively compensates for the inoperable status, restores the single failure capability and provides the required initiation capability of the instrumentation. Therefore, providing this option does not impact safety. However, if this action would result in system actuation, then declaring the system inoperable is the preferred action. Proposed Action A.2 includes a note that stipulates that it is "Not applicable if inoperable channel is the result of an inoperable breaker." This is acceptable since if a Recirculation Pump Trip (RPT) breaker is tripped it results in the tripping of the associated Recirculation Pump. The intent of Action A.2 as previously stated is to maintain single failure capability to assure initiation occurs upon a valid signal. Tripping the breaker performs the initiation without a valid signal and is not the desired action. These changes are consistent with the BWR Standard Technical Specifications NUREG-1433.

L3 A 2 hour Completion Time to restore one trip system to OPERABLE status if both trip systems are inoperable (i.e., restore EOC-RPT trip capability) has been added (Required Action B.1), to be consistent with the time provided in LCO 3.2.2 to restore MCPR. (The instrumentation is provided to protect the MCPR limit from being exceeded). CTS Table 3.2.B, Note 17 requires an orderly power reduction to be initiated immediately. The change from an immediate action to initiate power reduction to allowing 2 hours to restore EOC-RPT trip capability prior to initiating power reduction is a less restrictive change. The 2 hours allows adequate time for restoration of the EOC-RPT trip capability by the operator. The restoration of EOC-RPT trip capability will preclude plant maneuvers to reduce power which results in less opportunity to commit errors. The likelihood of an event requiring actuation of EOC-RPT during the two hours allowed to return the instrumentation to service is statistically small. CTS LCO 3.5.K allows two hours to return MCPR to prescribed limits when it is determined that the limiting value is being exceeded. Thus it is concluded that since the change is consistent with other LCOs to return MCPR to within limits, the change reduces possibility of errors during plant maneuvers, and the risk is small it is concluded that the change is acceptable. Also the changes are consistent with the BWR Standard Technical Specifications, NUREG-1433.

L4 The purpose of this instrumentation is to ensure an MCPR Safety Limit Violation will not occur late in core life due to a turbine trip or generator load rejection. LCO 3.3.4.1.a addresses the instrumentation requirements for the EOC-RPT to perform its required function. A new LCO (3.3.4.1.b) is proposed as an alternative. Since the instrumentation protects against a MCPR SL violation, with the instrumentation inoperable, modifications to the MCPR limits (LCO 3.2.2) may be applied to allow this LCO to be met. The MCPR penalty for the EOC-RPT inoperable condition is specified in the COLR.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.5.1 - ECCS INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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. In the specific case of the ECCS instrumentation and Limiting Safety Setting Sections that list ECCS System instrumentation setpoints, the Specifications have been combined into one Specification.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 The column title is now on a Per Function basis rather than the current Per Trip System basis. Thus, in general, the number of channels in the proposed columns is doubled since most functions have two trip systems. This new categorization is used for all ECCS, except the ADS. For the ADS, each of the two trip systems generally are listed in the Table, thus, the channels Per Function generally do not change. Exceptions being for ADS Functions 4.e, 4.f, 5.e, and 5.f where the proposed number of channels will be based on the number of pump pressure switches and for Functions 4.g and 5.g where the number of channels will be based on the number of timers.

A3 The Unit 1 CTS calibration frequencies for these functions are not consistent with Units 2 and 3. These frequencies have been reflected in the proposed BFN ISTS for Unit 1 as the same as those in the proposed ISTS for Units 2 and 3. The Unit 1 Calibration frequencies for these functions will be validated prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A4 The provisions of this note were used during the Unit 2, Cycle 7 outage and are no longer applicable. As such, they are being deleted.

A5 These proposed changes provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry is allowed for each . . . ") provides direction consistent with the intent of the existing Action for

an inoperable ECCS instrumentation channel. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative.

A6 Frequencies have been changed "from once/operating cycle" to "18 months" and from "once/day to once/24 hours. These changes are administrative since the current and revised frequencies are the same.

A7 The manner in which the number of required channels is described has been changed to a per pump designation. This change is administrative since the current interpretation assumes one trip system per pump, as applicable, for these Functions. CS has one timer per pump with normal and diesel power. Units 1 and 2 LPCI has two timers on two pumps (C & D on Unit 2 and A & B on Unit 1) and one timer on two pumps (A & B on Unit 2 and C & D on Unit 1) with normal and diesel power. Unit 3 LPCI has two timers on each of the four pumps with normal and diesel power. Footnote (e) has been added to clarify the required number of channels.

A8 For Specification 3.3.5.1, only notes 1, 4, 6, 7, 27 (for Units 1 and 2), 28, and 29 (for Unit 3) of Current Technical Specification Table 4.2.A apply. The remaining Notes will be addressed as appropriate in other applicable Specifications.

A9 The ADS Instrumentation has been divided into two parts, Functions 4 and 5, with Function 4 being the ADS Trip System A and Function 5 being the ADS Trip System B. No technical changes are made.

A10 Deleted last paragraph of TS 4.2.B which clarifies that whenever a system or loop is made inoperable because of a required test or calibration, the other systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect that they are inoperable. This statement provides no specific information and is only clarifying in nature. Therefore, its deletion is considered an administrative change.

A11 The current Applicability for the ECCS Instrumentation is whenever the CSCS system(s) is required to be OPERABLE as specified in Section 3.5. The changes to the specific ECCS System Applicabilities were described in the Justification for Changes for Section 3.5. This proposed change specifies by a footnote (footnote d) that the only time the HPCI and ADS Functions are required to be Operable in Modes 2 and 3 is with reactor steam dome pressure > 150 psig. Since the Applicability of the HPCI and ADS Instrumentation is consistent with the requirements of the HPCI System and ADS Specifications in Section 3.5, this is considered an administrative change. This change is consistent with NUREG-1433.

A12 Current Technical Specifications provide separate actions and SRs for ECCS initiation logic. In ISTS, Actions and SRs for the logic functions are captured by the Actions and SRs for the Initiation Functions. Justifications for Changes to Actions and SRs for the individual initiation functions have been provided as appropriate. Proposed SRs 3.3.5.1.2 and 3.3.5.1.6, which require that functional and logic system functional testing be performed on a periodic basis, capture all operability testing requirements for these logic functions. Therefore, the deletion of these logic functions is considered administrative.

A13 CTS Table 3.2.B lists 1 channel as required per trip system for the Condensate Header Low Level and Suppression Chamber High Level Functions (Proposed ITS 3.3.5.1 Functions 3.d and 3.d). CTS Table 3.2.B lists 2 channels as required for Reactor Vessel Water Level - High, Level 8 Function (Proposed ITS Function 3.c). CTS Note 2 to the Table for 3.2.B states these functions only have one trip system. Therefore, the required channels per function are one for Condensate Header Low Level and Suppression Chamber High Level Functions and two for the Reactor Vessel Water Level - High, Level 8 Function.

TECHNICAL CHANGES - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These Requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 Proposed Functions 1.d and 3.f have been added. Function 1.d is assumed to be operable and capable of closing the minimum flow valve to ensure the assumed ECCS flow during transient and accident analyses are met. Both Functions protect the associated ECCS pumps from overheating when the pump is operating and the associated injection valve is not open. These instruments currently exist but are not a Technical Specification requirement. Therefore, their addition is considered an additional restriction on plant operation.

M2 Deleted.

M3 MODE 4 and 5 requirements have been added, since LPCI and CS now have requirements in these MODES (see Justification for Changes to ISTS 3.5.2). These additions are additional restrictions on plant operation.

M4 Current Technical Specification Table 3.2.B, Note 16, explicitly allows one ADS trip system to be taken out of service for functional or calibration testing for a period not to exceed 8 hours. Proposed ISTS 3.3.5.1 SR Note 2 has similar provisions, however, the ISTS allows only 6 hours. The reduction in time is an additional restriction on plant operation.

M5 CTS Table 3.2.B, Note 19 was added on April 16, 1996, by Amendments 229, 244, and 204 for Units 1, 2 and 3 respectively to allow testing of the reactor coolant system instrument line flow check valves provided that manual and automatic initiating capability of Core Spray and LPCI are maintained and the reactor is in Cold Shutdown. This was necessary since testing of these check valves resulted in the inability to meet Note 1 to Table 3.2.B for affected instrumentation. Required Action B.1 and B.3 of proposed Specification 3.3.5.1 will allow this testing since a channel can be inoperable for 24 hours and at that point the channel can be placed in trip. Therefore, the provision of Note 19 is no longer required.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 System design and operational details have been relocated to the Bases and procedures. Trip setpoints are an operational detail not directly related to the operability of the instrumentation. The Allowable Value is the required limitation of the parameter and this value is retained. The CTS "Trip Level Settings" are equivalent to ITS "Allowable Values." TVA's methodology for determination of setpoints utilizes the CTS "Trip Level Settings" as the allowable value in establishing the nominal trip setpoint.

The selection of nominal trip setpoints plus associated inaccuracies ensures the CTS "Trip Level Settings" are not exceeded. TVA's setpoint methodology is consistent with RG 1.105 which endorses ISA Standard ISA-S67.04-1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants" and has been reviewed by the NRC in previous submittals, e.g. NRC Letter to Mr. Oliver D. Kingsley dated January 2, 1991, Issuance of Amendment (TAC No. 77279) (TS291). Details relating to system design and operation (e.g., bypasses, associated division, specific equipment affected) are also unnecessary in the LCO and have been relocated to the Bases and procedures. The design features and system operation are also described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by provisions of 10 CFR 50.59.

LA2 Details of the methods for performing surveillances are relocated to the Bases and procedures. The design features and system operation which dictate the methods are described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

LA3 Recategorized as a Relocation. Reference Justification for Change R3.

LB1 An allowed out of service time (AOT) for testing of 6 hours has been added. SR Note 2 allows 6 hours for testing prior to entry into the LCO. The AOT for repair, which allows placing CS, LPCI, and HPCI channels in trip or restoring the channel to OPERABLE status, remains at 24 hours for all Functions except the minimum flow Functions, and is extended to 7 days for the minimum flow Functions. The allowed out of service time for placing ADS channels in trip, or to restore the channel to OPERABLE status, is extended to 96 hours if HPCI or RCIC is also inoperable, or 8 days if both HPCI and RCIC are OPERABLE. The channel functional test frequency (STI) has been extended to once per 92 days from monthly. These AOTs and STIs have been shown to maintain an acceptable risk in accordance with previously conducted reliability analyses (NEDC-30936-P-A, December 1988).

"Specific"

L1 Proposed Required Actions B.3, and F.2 have been added to allow an inoperable channel to be placed in the tripped condition rather than declaring the associated supported feature inoperable. This conservatively compensates for the inoperable status, restores the single failure capability and provides the required initiation capability of the instrumentation. Therefore, providing this option does not impact safety. However, if this action would result in system actuation, then declaring the system inoperable is the preferred action.

L2 Proposed Note (c) to proposed Table 3.3.5.1-1 modifies the MODES 1, 2, and 3 requirements for the Reactor Vessel Steam Dome Pressure - Low Function to only be required when the associated recirculation pump discharge valve is open. With the valve closed, the instrument's function has been completed. Re-opening of the valve is a controlled evolution, and is not performed without strict administrative controls.

L3 CTS Table 3.2.B, Note 1.B requires a system or component to be declared inoperable if the same function is inoperable in more than one trip system or the first column is reduced by more than 1. Proposed BFN ISTS 3.3.5.1, Required Actions B.1, B.2, C.1, D.1, E.1, F.1, and G.1 provide a one hour Completion Time from discovery of loss of initiation capability to make this declaration. The proposed actions do not

require the system or component to be declared inoperable unless initiation capability is lost. This is acceptable since the Function is still capable of performing its design function. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels.

RELOCATED SPECIFICATIONS

R1 Trip System Bus Power Monitors and Core Spray Sparger Differential Pressure functions are operational functions only and are not considered in any design basis accident or transient. The evaluation summarized in the Browns Ferry Unit 1, 2, and 3 split report determined the loss of these Functions to be a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for these functions are being relocated to the Technical Requirements Manual. Any change to these instrument functions will be controlled by the provisions of 10 CFR 50.59.

R2 This instrument Function is being relocated to plant specific controls. The purpose of this instrument is to preclude inadvertent actuation of drywell and suppression pool sprays during a LOCA. If a LOCA signal is present, the drywell and suppression pool spray valves cannot be opened unless reactor vessel water level is above the 2/3 core height level (to preclude diversion of LPCI when it is needed for core flooding) and the drywell pressure is $\geq$ 1.0 psig and $\leq$ 2.5 psig (indicative of a valid need for operating the drywell and suppression pool sprays). If the instrument is inoperable such that it trips too soon or too late (or not at all), the LPCI System is not impacted.

If the instrument trips too soon, the reactor vessel water level 2/3 core height Function still ensures that flow is not diverted away from core flooding. In fact, the major contributor to potential flow diversion is suppression pool cooling, and its valves are only precluded from opening by the 2/3 core height instrument. The flow diverted by the drywell and suppression pool sprays is a small fraction of that diverted by suppression pool cooling. Thus, operability of LPCI is not impacted. While tripping of the instrument allows one of the permissives for opening drywell and suppression pool spray valves to be met, inadvertent operation does not result, since manual actions must still be taken to open the valves if the other permissive (2/3 core height) is also met. In addition, if a LOCA signal is not present, this instrument does not preclude operation of the drywell and suppression pool spray valves. Therefore, inadvertent operation of drywell spray has been analyzed at BFN and does not result in containment failure due to operation of the reactor building-to-suppression chamber and the

suppression chamber-to-drywell vacuum breakers. These vacuum breakers are controlled by Technical Specifications (current and proposed). Therefore, operability of the Suppression Pool Spray System is not impacted.

If the instrument trips too late or not at all, then no flow can be diverted by the drywell and suppression pool sprays; thus LPCI is not affected. The only Technical Specification systems affected in this case are the Suppression Pool Spray and the Drywell Spray Systems. A failure of the instrument to function would preclude the suppression pool spray and drywell spray valves from being opened from the control room. However, these systems are manually controlled systems that are not needed for a minimum of 10 minutes following a DBA LOCA, and the valves could still be opened locally at the valve operator. In addition, the instrument could be overridden to allow operation from the control room. Therefore, failure of this instrument may not result in the Suppression Pool Spray or Drywell Spray Systems being inoperable.

Since this instrument does not relate to LPCI Operability, and the Suppression Pool Spray and Drywell Spray Systems are manually actuated systems, this instrument Function is being relocated to the Technical Requirements Manual. Any change to this instrument function will be controlled by the provisions of 10 CFR 50.59.

R3 Core Spray Loop A & B Discharge Pressure, RHR Loop A & B Discharge Pressure, RHR and CS Cooler Fan Logic, RHRSW Start on CS Start, Instrument Channel - Thermostat (RHR Area Cooler Fan), and Instrument Channel - Thermostat (Core Spray Area Cooler Fan) are operational functions only and are not considered in any design basis accident or transient. As such, they are being relocated to the Technical Requirements Manual. Relocating requirements for these instrument channels does not preclude them from being maintained operable. They are required to be operable in order to support LPCI and CS system operability. If they become inoperable, the operability of the supported systems are required to be evaluated under the Safety Function Determination Program in Section 5.0 of the Technical Specifications. This change is consistent with NUREG-1433.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.6.1 - PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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. In the specific case of the PCIS instrumentation and Limiting Safety Setting Sections that list PCIS instrumentation setpoints, the Specifications have been combined into one Specification.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 This action has been modified to be in MODE 2, instead of to close the MSIVs. This is essentially the same, since to close the MSIVs, the unit must be in MODE 2. Once in MODE 2, the Function is not required (as stated in the remarks section for the Function), thus, the MSIVs are not required to be closed. Therefore, this change is considered administrative.

A3 For Specification 3.3.6.1, these notes (2, 3, 4, 5 (excluding Unit 1), 7, 8, 12, 13, 14, 15, 16, 17, 19, 20, 22, 23, 24, 25, 26, 30, 32) for Current Technical Specification Tables 4.2.A and 4.2.B are not used. These Notes will be addressed as appropriate in other applicable Specifications.

A4 The format of the proposed Technical Specifications does not include providing "cross-references." LCO 3.0.8 adequately prescribes the use of special operations LCOs without such references. Therefore, the existing references to Special Test Exceptions serve no functional purpose, and their removal is purely an administrative difference in presentation.

A5 The Frequency "once/operating cycle" and "during outage only" has been changed to "18 months". The Frequency "once/3 months" has been changed to "92 days". The Frequency "once/4 months" has been changed to "122 days". The Frequency of "once/day" has been changed to "24 hours." These are changes in nomenclature and are considered administrative.

A6 Deleted last paragraph of TS 4.2.B which clarifies that whenever a system or loop is made inoperable because of a required test or calibration, the other systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect that they are inoperable. This statement provides no specific information and is only clarifying in nature. Therefore, its deletion is considered an administrative change.

A7 CTS Table 3.2.A, Note 1.D requires the affected system be declared inoperable after closing the affected system's isolation valves (RHR shutdown cooling) as required by the CTS Action for having less than the minimum channels operable required for the high drywell pressure function. This is an unnecessary statement and has been deleted. An Action requiring the affected system to be declared inoperable is an unnecessary reminder that other Technical Specifications may be affected.

A8 These proposed changes provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry is allowed for each . . .") and "one or more automatic Functions" provides direction consistent with the intent of the existing Action for an inoperable isolation instrumentation channel. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative.

A9 Current Technical Specifications provide separate Actions for initiating instruments, initiation logic, and actuation logic. The ISTS only specifies Actions for initiating instruments. The Actions for the initiation logic are included in the Actions for the initiating instruments. For example, Group 2 initiation logic is actuated by Reactor Vessel Low Water Level or High Drywell Pressure. The CTS Actions for Reactor Vessel Low Water Level and High Drywell Pressure are A or (B and E) which is the same as the Action for Group 2 initiation logic. Therefore, the deletion of the initiation logic is considered administrative. Failure of actuation logic would make the associated Primary Containment Isolation Valve inoperable and result in ITS Section 3.6.1.3 Actions. The less restrictive changes to the Actions for the actuation logic versus the Actions required for an inoperable valve are addressed in L8, and the more restrictive changes to the Actions for the actuation logic versus the Actions required for an inoperable valve are addressed in M12.

A10 CTS Table 3.2.B, Note 8, which states that Note 1 does not apply to this item, has been deleted since the note is not referenced from the Table.

A6 Deleted last paragraph of TS 4.2.B which clarifies that whenever a system or loop is made inoperable because of a required test or calibration, the other systems or loops that are required to be OPERABLE shall be considered OPERABLE if they are within the required surveillance testing frequency and there is no reason to suspect that they are inoperable. This statement provides no specific information and is only clarifying in nature. Therefore, its deletion is considered an administrative change.

A7 CTS Table 3.2.A, Note 1.D requires the affected system be declared inoperable after closing the affected system's isolation valves (RHR shutdown cooling) as required by the CTS Action for having less than the minimum channels operable required for the high drywell pressure function. This is an unnecessary statement and has been deleted. An Action requiring the affected system to be declared inoperable is an unnecessary reminder that other Technical Specifications may be affected.

A8 These proposed changes provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry is allowed for each . . .") and "one or more automatic Functions" provides direction consistent with the intent of the existing Action for an inoperable isolation instrumentation channel. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative.

A9 Current Technical Specifications provide separate Actions for initiating instruments, initiation logic, and actuation logic. The ISTS only specifies Actions for initiating instruments. The Actions for the initiation logic are included in the Actions for the initiating instruments. For example, Group 2 initiation logic is actuated by Reactor Vessel Low Water Level or High Drywell Pressure. The CTS Actions for Reactor Vessel Low Water Level and High Drywell Pressure are A or (B and E) which is the same as the Action for Group 2 initiation logic. Therefore, the deletion of the initiation logic is considered administrative. Failure of actuation logic would make the associated Primary Containment Isolation Valve inoperable and result in ITS Section 3.6.1.3 Actions. The less restrictive changes to the Actions for the actuation logic versus the Actions required for an inoperable valve are addressed in L8, and the more restrictive changes to the Actions for the actuation logic versus the Actions required for an inoperable valve are addressed in M12.

A10 CTS Table 3.2.B, Note 8, which states that Note 1 does not apply to this item, has been deleted since the note is not referenced from the Table.

A11 Deleted.

A12 The RWCU Temperature functions in the CTS for Unit 1 are not consistent with those listed for the CTS for Units 2 and 3. However, the proposed BFN ISTS for Unit 1 has been made consistent with Units 2 and 3. The differences in these functions will be resolved prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A13 The Unit 1 CTS calibration frequencies for these functions are not consistent with Units 2 and 3. These frequencies have been reflected in the proposed BFN ISTS for Unit 1 as the same as those in the proposed ISTS for Units 2 and 3. The Unit 1 Calibration frequencies for these functions will be validated prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A14 The Unit 1 channel check requirements for these functions are not consistent with Units 2 and 3. These channel checks have been reflected in the proposed BFN ISTS for Unit 1 as the same as those in the proposed ISTS for Units 2 and 3. The Unit 1 channel check requirements for these functions will be validated prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A15 The HPCI and RCIC Steam Line Space Temperature Functions have been changed for clarification to identify each location monitored as a separate Function and each sensor as a channel (similar to Functions for RWCU isolation). The change is considered administrative since the number of required instruments are not changed. The CTS Bases states high temperature in the vicinity of the HPCI equipment is sensed by four sets of four bimetallic temperature switches. The 16 temperature switches are arranged in two trip systems with eight temperature switches in each trip system. The CTS Bases also states the RCIC high area temperature sensing instrument channels are arranged in the same manner as the HPCI system. The Unit 1 HPCI Steam Line Space Temperature Function Areas are different than the Unit 2 and 3 areas. However, the proposed BFN ISTS for Unit 1 has been made the consistent with Units 2 and 3. The differences in these functions will be resolved prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A16 The proposed ISTS addresses the BFN Group 2 isolation valves in two separate sets of Functions (proposed ISTS Function sets 2 and 6). The proposed reference Condition in Table 3.3.6.1-1 for the Drywell Pressure - High Function of the Shutdown Cooling (SDC) System Isolation Function set (proposed Function 6.c) is Condition F, which requires the affected penetration flow paths to be isolated in one hour. This change is considered to be administrative since the CTS referenced action in CTS Table 3.2.A for inoperable Group 2 (RHR Isolation Actuation) Logic is

CTS Action D, which similarly requires the affected penetration flow paths to be isolated in one hour.

A17 Deleted (Per NRC Question 3.3.6.1-21).

A18 The MSL High Flow Function in CTS Table 3.2.A is listed with two channels as required to be OPERABLE per trip system. The proposed ISTS 3.3.6.1 Function 1.c required channels per trip systems is two per MSL, which is based on each channel containing one sensor and two PCIS trip systems (two channels per MSL per trip system X four MSL X two trip systems equals 17 total channels). This change is considered administrative since CTS Table 3.2.A, Note 3 states that there are four sensors per MSL (four sensors per MSL X four MSL equals 16 total sensors). Similarly, the MSL Tunnel High Temperature Function in CST Table 3.2.A is listed with two channels as required to be OPERABLE per trip system. The proposed ISTS 3.3.6.1 Function 1.d required channels per trip system is eight, which is based on each channel containing one sensor. This change is considered administrative since CTS Table 3.2.A, Note 12 clarifies the CTS channel bases and identifies the CTS considers a channel to contain four sensors.

A19 CTS Notes for Tables 4.2.A through 4.2.L except 4.2.D and 4.2.K, Notes 18, 21, 27 (Units 1 and 2), 28, and 29 (Unit 3), have been deleted. CTS Notes 27 and 28 for Units 1 and 2 and Notes 28 and 29 for Unit 3 provide a general description of channel calibration and channel functional test. The deletion of these notes are considered administrative since the requirements of these notes are enveloped by the proposed ISTS Definitions for CHANNEL CALIBRATION and CHANNEL FUNCTIONAL TEST. CTS Notes 18 and 21 identify limitations on conditions for surveillance performance (e.g. surveillance is limited to conditions where operation of the equipment is permissible). The deletion of these notes are considered administrative since limitations on equipment operation are addressed by proposed ISTS for the actuated features and proposed ISTS Section 3.0 (i.e., device operation resulting in an LCO not being met would result in entry into the Required Actions of the LCO per proposed ISTS LCO 3.0.2).

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These Requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based

on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 An additional action, to be in MODE 3 within 12 hours, has been added to those requirements which require a shutdown to MODE 4. These times are consistent with the BWR Standard Technical Specifications, NUREG 1433, and are additional restrictions on plant operation.

M2 Not used.

M3 The CTS Action for Functions 2.a and 2.b and the corresponding Group 2 initiating logic is to take Table 3.2.A, Note 1.A actions (initiate an orderly shutdown . . .) or Note 1.B actions (to initiate an orderly load reduction and have the MSLs isolated) and Note 1.E actions (to initiate primary containment isolation within 24 hours). The proposed Action for these Functions, which is consistent with the BWR/4 Standard Technical Specifications, NUREG-1433, does not include the latter option. Therefore, the proposed change is more restrictive.

M4 Not used.

M5 Not used.

M6 The Action for an inoperable channel for the Main Steam Line Pressure - Low Function in Existing Table 3.2.A is to isolate the MSL within 8 hours. Under the same conditions, proposed Specification 3.3.6.1-1 (Table 3.3.6.1-1, Function 1.b, Condition E) will require that the reactor be in MODE 2 within 6 hours. This change is acceptable because it places the reactor outside the Mode of Applicability in less time than the current Specifications. This change consistent with the BWR/4 Standard Technical Specifications, NUREG-1433.

M7 Not used.

M8 Proposed Action F.1 requires the licensee to isolate the affected penetration flow path for these HPCI and RCIC functions, rather than just declare the system or component inoperable (CTS Table 3.2.B, Note 1.B) when the isolation capability can not be maintained. In the case of isolation capability, it is more appropriate to isolate the affected penetration, at which time the system would be inoperable. Regardless, isolating the affected penetration effectively renders the associated system inoperable and the appropriate LCO would be entered.

M9 CTS 3.2.A requires primary containment isolation instrumentation to be operable when primary containment integrity is required. Per CTS 3.7.A.2.a, primary containment integrity is required to be OPERABLE at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. The proposed BFN ISTS 3.3.6.1 applicability for most functions is MODES 1, 2, and 3. This is more restrictive since CTS does not require the primary containment integrity when in MODE 2, not critical and < 212°F.

M10 Function 5.g, SLC system initiation, has been added. This function ensures the safety analysis assumptions are met. Appropriate ACTIONS and Surveillance Requirements have also been added. This is an additional restriction on plant operation. Proposed Note (a) to Table 3.3.6.1-1 clarifies that one SLC System initiation signal provides logic input to close both RWCU valves.

M11 The RHR Shutdown Cooling low water level isolation Function (proposed ISTS 3.3.6.1 Function 6.b) applicability has been revised to add operability requirements in Modes 4 and 5, since this is when it is needed to isolate if an inadvertent drain down event occurred. In the CTS this Function is not required to be OPERABLE in Modes 4 and 5 (see DOC M9). Proposed ISTS Note (b) will require only one channel per trip system to be OPERABLE in Modes 4 and 5 when RHR SDC system integrity is maintained. The actions for this function have been revised to require action to be immediately initiated to isolate the affected lines or to restore the channels to OPERABLE status. These new actions ensure that SDC is not interrupted when needed, yet also ensures appropriate actions are continued. These changes are consistent with the BWR Standard Technical Specifications, NUREG-1433, and are additional restrictions on plant operation.

M12 For Primary Containment Isolation Instrumentation the CTS required action for inoperable Group 6 Logic, including Group 2 (Reactor Building & Refueling Floor, and Drywell Vent and Purge Actuation) portions of the Group 6 Logic, has been changed from suspending handling of spent fuel, operations over spent fuels, and open reactor wells (CTS Table 3.2.A, Note 1.F) and isolating the reactor building and starting the Standby Gas Treatment System (CTS Table 3.2.A, Note 1.G) to isolating the affected penetration flow path(s) in one hour (proposed ISTS 3.3.6.1, Required Action F.1). Due to instrument commonalties, CTS Table 3.2.A addresses both Primary Containment Isolation Instrumentation and Secondary Containment Isolation Instrumentation, and thus, provides required actions based on considerations for loss of both features. In the proposed ISTS separate specification are provided for Primary Containment Isolation Instrumentation (proposed ISTS 3.3.6.1) and Secondary Containment Isolation Instrumentation (proposed ISTS 3.3.6.2). Since portions of this logic are shared with the Secondary Containment Isolation Instrumentation, when inoperable logic components affect

Secondary Containment Isolation, the Required Actions of proposed ISTS 3.3.6.2 are entered which require actions similar to the CTS (changes from CTS Table 3.2.A, Note 1.F and 1.G are addressed in DOCs for proposed ISTS 3.3.6.2). The proposed change is considered more restrictive since when the common logic components are inoperable, in addition to actions relating to Secondary Containment (required by proposed ISTS 3.3.6.2 as discussed above), the proposed change will require the affected primary containment penetration flow path(s) to also be isolated. The proposed change is acceptable since isolating the affected penetration flow path(s) accomplishes the safety function of the inoperable logic and results in actions that are consistent with the type of actions in NUREG-1433 for inoperable Primary Containment Isolation Instrumentation.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 Information describing system design and purpose which is not needed to define the Technical Specification operability requirements has not been included in the proposed ISTS. This information is only a description of the facility and this type of information is typically provided in the FSAR. Changes to the facility and the FSAR will be controlled by the provisions of 10 CFR 50.59. Instrument identifiers (UNIDs) have not been included in the proposed ISTS, but are included in the proposed ISTS Bases as appropriate. Trip setpoints are an operational detail not directly related to the operability of the instrumentation. The Allowable Value is the required limitation of the parameter and this value is retained. The CTS "trip level settings" are equivalent to ISTS "Allowable Values." TVA's methodology for determination of setpoints utilizes the CTS "trip level settings" as the allowable value in establishing the nominal trip setpoint. The selection of nominal trip setpoints plus associated inaccuracies ensures the CTS "trip level settings" are not exceeded. TVA's setpoint methodology is consistent with RG 1.1065 which endorses ISA Standard ISA-S67.04-1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants" and has been reviewed by the NRC in previous submittals, e.g., NRC letter to Mr. Oliver D. Kingsley dated January 2, 1991, Issuance of Amendment (TAC No. 77279)(TS291).

LA2 The requirement to include a calibration once per operating cycle of time delay relays and timers necessary for proper functioning of the logic in the logic system functional tests (CTS Notes for Tables 4.2.A through 4.2.L except 4.2.D and 4.2.K, Table 4.2.B, Note 6) has been relocated to the proposed ISTS Bases for SR 3.3.6.1.6.

LA3 CTS Table 3.2.A, Note 12 contains compensatory actions associated with

recovery of a loss of ventilation in the MSL tunnel. These compensatory actions are not needed to satisfy Required Actions for a complete loss of isolation function specified in NUREG-1433 but represent good engineering practice. Therefore, the compensatory actions associated with recovery of a loss of ventilation in the MSL tunnel currently in Note 12 are being relocated to the Bases.

LA4 Identification that the HPCI and RCIC Steam Supply Pressure - Low and Turbine Exhaust Diaphragm Pressure - High Functions have only one trip system (CTS Table 3.2.B, Note 2) has been relocated to the proposed ISTS Bases. The bases for the functional test frequency of the Low Pressure Main Steam Line and High Flow Main Steam Line Functions are given in CTS Table 4.2.A, Note 29 for Units 1 and 2 and Note 27 for Unit 3 as per NUREG-0737. The bases for the functional test frequency of the Low Pressure Main Steam Line and High Flow Main Steam Line Functions have been relocated to the proposed ISTS Bases for SR 3.3.6.1.2 and changed to the reliability analysis described in NEDC-31677P-A.

LB1 The allowed out of service time (AOT) for placing a channel in trip when one trip system has inoperable, untripped channels, is extended to 12 hours for those channels common to RPS and 24 hours for all other channels. The AOTs for placing a channel in trip for required surveillance testing have been extended from 4 hours to 6 hours. The channel functional test frequency (STI) has been extended to once per 92 days from monthly. These AOTs and STIs have been shown to maintain an acceptable risk in accordance with previously conducted reliability analysis (NEDC-30851-P-A, Supplement 2, March 1989 and NEDC-31677-P-A, July 1990).

"Specific"

L1 For HPCI/RCIC primary containment isolation functions, CTS Table 3.2.B, Note 1.B requires the supported system or component to be declared inoperable if the same function is inoperable in more than one trip system or the first column of Table 3.2.B is reduced by more than 1. For the remaining isolation functions, CTS Table 3.2.A, Note 1, requires the appropriate action listed in Table 3.2.A be taken if, for the same function, the first column of Table 3.2.A cannot be met for all trip systems. Proposed BFN ISTS 3.3.6.1, Required Action B.1 provides a one hour Completion Time from discovery of loss of isolation capability to make this declaration. Proposed Required Action B.1 does not require further action unless isolation capability is lost. This is acceptable since the Function is still capable of performing its design function. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in isolation capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels.

L2 The time to close the MSIVs has been extended from 8 hours to 12 hours, and the time to reach MODE 4 (Cold Shutdown) has been extended from 24 to 36 hours. This provides the necessary time to close the MSIVs or shutdown the plant in a controlled and orderly manner that is within the capabilities of the unit, assuming the minimum required equipment is OPERABLE. This extra time reduces the potential for a unit upset that could challenge safety systems. This time is consistent with the BWR Standard Technical Specifications, NUREG-1433.

L3 Mode 1 and 2 applicability for isolation of the RHR Shutdown Cooling valves by the available low water level signal (proposed ISTS 3.3.6.1 Function 6.b) has not been included in the proposed ISTS. The RHR Shutdown Cooling Supply isolation valves are maintained isolated by the Reactor Vessel High Pressure signal (proposed ISTS 3.3.6.1 Function 6.a in Modes 1, 2, and 3) when there is significant pressure in the Reactor Vessel. Both the RHR Shutdown Cooling Supply and the RHR LPCI to Reactor isolation valves receive isolation signals from the Drywell Pressure - High Function (proposed ISTS 3.3.6.1 Function 6.c) which is required to be OPERABLE in Modes 1 2, and 3. The RHR LPCI to Reactor isolation valves are essential for post-accident mitigation, and therefore, the PCIS isolation signal to these valves are only enabled when in Shutdown Cooling (both RHR Shutdown Cooling Supply isolation valves open). Based on these considerations and administrative controls which prevent unexpected loss of inventory via these flow paths, the deletion of the Mode 1 and 2 applicability for isolation of the RHR Shutdown Cooling valves by the available low water level signal (proposed ISTS 3.3.6.1 Function 6.b) is acceptable. This change is consistent with the BWR Standard Technical Specifications, NUREG-1433.

L4 The Required Action for when the RWCU low water level isolation Function is inoperable has been modified to allow the valves to be isolated in 1 hour, instead of requiring a unit shutdown. Isolation of the affected line returns the system to a status where it has performed its function, thus continued operation should be allowed.

L5 The Required Action for the Main Steam Line Functions (excluding proposed Function 1.b) is proposed to allow isolation of the affected penetration flow path (currently a shutdown is required for Reactor Vessel Water Level - Low Low Low (Level 1) and Group 1 (Initiating Logic)). Some conditions may affect the isolation logic for only one penetration flow path. In these cases, it is not necessary to require a shutdown of the unit; rather, isolation of the affected line returns the system to a status where it can perform the remainder of its isolation function, and continued operation is allowed (although it may be at a reduced power level in MODE 2.)

because no time is specified in the CTS but is acceptable because CTS 3.7.D.2 allows four hours to isolate the line if the valve becomes inoperable which has the same effect as inoperable Actuation Logic for the valve.

The CTS Table 3.2.A Action for Group 2 (Drywell Sump Drains - Actuation) Logic is K "Manually isolate the affected lines." If the Actuation Logic failure made one valve in a penetration flow path inoperable then proposed ITS LCO 3.6.1.3 Action A.1 would require one valve in the flow path to be isolated within four hours and if it made two valves in a penetration flow path inoperable then LCO 3.6.1.3 Action B.1 would require the one valve in the flow path to be isolated within 1 hour. The ITS Action is less restrictive than the CTS Action because no time is specified in the CTS but is acceptable because CTS 3.7.D.2 allows four hours to isolate the line if one valve becomes inoperable which has the same effect as inoperable Actuation Logic for the valve.

The CTS Table 3.2.A Action for Group 3 (Actuation) Logic is C "Isolate Reactor Water Cleanup System." If the Actuation Logic failure made one valve in a penetration flow path inoperable then proposed ITS LCO 3.6.1.3 Action A.1 would require one valve in the flow path to be isolated within four hours and if it made two valves in a penetration flow path inoperable then LCO 3.6.1.3 Action B.1 would require the one valve in the flow path to be isolated within one hours. The ITS Action is less restrictive than the CTS Action because no time is specified in the CTS but is acceptable because CTS 3.7.D.2 allows four hours to isolate the line if one valve becomes inoperable which has the same effect as inoperable Actuation Logic for the valve.

The CTS Table 3.2.A Action for Functions 5.a, 5.b, 5.c, 5.d, 5.e, and 5.f is C "Isolate Reactor Water Cleanup System." The ISTS proposed Action is F.1 "Isolate the affected penetration flow path(s)" with a Completion Time of one hour. The proposed change is less restrictive since one hour is allowed to complete required Actions. The one hour Completion Time is acceptable because it minimizes the risk while allowing sufficient time for plant operations to isolate the affected penetration flow path(s).

The CTS Table 3.2.A Action for RCIC System (Isolation) Instrument Channels and HPCI System (Isolation) Instrument Channels is B "Declare the system or component inoperable." The proposed ITS Action to replace it is F "Isolate the affected penetration flow path(s)" with a Completion Time of "one hour." Isolating the penetration flow path is more restrictive as addressed in M8 but the one hour completion time is less restrictive as CTS do not specify a time. The one hour completion time is acceptable because it minimizes the risk while allowing sufficient time for plant operations to isolate the affected penetration flow path(s).

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.6.2 - SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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 done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 These proposed changes provide more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry is allowed for each . . .") and "one or more automatic Functions" provides direction consistent with the intent of the existing Action for an inoperable isolation instrumentation channel. Since this change only provides more explicit direction of the current interpretation of the existing specifications, this change is considered administrative.

A3 The Frequency "once/operating cycle" has been changed to "18 months." The Frequency "once/6 months" has been changed to "184 days." The Frequency of "once/day" has been changed to "24 hours." These are changes in nomenclature and are considered administrative.

A4 The last sentence of Note 8 to Table 4.2.A has been deleted since this allowance is specified in proposed SR 3.0.1.

A5 Deleted.

A6 Proposed ISTS 3.3.6.2 Required Actions C.1.2 and C.2.2 have been added for proposed ISTS 3.3.6.2 Function 3 to provide guidance if secondary containment isolation or starting the SGT System is not possible or desired (refer to DOC M2 for the addition of C.1.2 and C.2.2 for proposed ISTS 3.3.6.2 Functions 1, 2, and 4). Currently, reactor shutdown per CTS 1.C.1 would be required if the required action to isolate the reactor building and start the standby gas treatment system (CTS Table 3.2.A Note 1.G) could not be performed. Under the proposed ISTS with the SCIVs and Standby Gas Treatment System declared inoperable both proposed ISTS LCO 3.6.4.2 Condition C and LCO 3.6.4.3 Condition D would require the reactor to be shutdown. Specifically, proposed ISTS LCO 3.6.4.3 Required Action D.1 would require proposed ISTS LCO 3.0.3 to be entered immediately. Changes in the generic shutdown LCO (CTS 1.C.1 changes relative to the proposed ISTS LCO 3.0.3) are addressed in DOCs for proposed ISTS Section 3.0. Since both the CTS and proposed ISTS would require reactor shutdown, the addition of the actions of proposed ISTS 3.3.6.2, Required Actions C.1.2 and C.2.2 are considered administrative for proposed ISTS 3.3.6.2 Functions 3. However, the added one hour completion time for declaring the equipment inoperable is less restrictive and is addressed in DOC L1.

A7 For Specification 3.3.6.2, these notes (2, 3, 5 (Units 2 and 3), 7, 11, 12, 13, 14, 15, 16, 17, 20, 21, 22, 23, 24, 25, 26, 27 (Unit 3), 29 (Units 1 and 2), 30 (Unit 2), 31, 32 (Units 1 and 3) for Current Technical Specification Table 4.2.A are not used. These Notes will be addressed as appropriate in other applicable Specifications.

A8 Current Technical Specifications provide separate actions and SRs for Secondary Containment Isolation Instrumentation initiating and isolation logic. In ISTS, Actions and SRs for the logic functions are captured by the Actions and SRs for the channels and Functions. When a logic system is discovered to be inoperable, the proposed LOGIC SYSTEM FUNCTIONAL TEST Surveillance Requirement is not met, and thus, the proposed ISTS LCO is not met per proposed SR 3.0.1. The proposed Bases for the LOGIC SYSTEM FUNCTIONAL TEST Surveillance Requirements state that these tests demonstrate the OPERABILITY of the required isolation logic for a specific channel. Thus, each channel which cannot perform the required initiation function through the logic system is declared inoperable for the associated feature and the proposed ISTS Conditions and Required Actions for the inoperable channels are entered. For example, when one of the three SGT Train Logic Systems is inoperable, all of the channels for proposed ISTS 3.3.6.2 Functions 1, 2, 3, and 4 would be declared inoperable for the associated SGT subsystem since all of the channels input through logic to the inoperable SGT Train Logic System. Under these conditions proposed ISTS Required Action A.1 would require the logic system to be restored to OPERABLE status in 12 hours based on the 12 hour Completion Time for Functions 1 and 2 and that in affect the channels cannot be placed in trip. Under these conditions proposed ISTS

Condition B would not be entered since the Functions would still be maintaining initiation capability for two SGT subsystems. If the inoperable SGT Train Logic System was not restored to OPERABLE status in 12 hours, proposed ISTS Condition C would be entered which would require in one hour the associated SGT subsystem to be placed in operation or declared inoperable If in the above example two of the three SGT Train Logic Systems were inoperable, the required actions would be the same except that proposed ISTS Condition B would be entered which would require in one hour at least one of the inoperable SGT Train Logic Systems to be restored to OPERABLE status or Condition C entered. Since actions and SRs are captured for the logic systems in the proposed ISTS as discussed above, the deletion of the listing of these logic functions is considered administrative. Changes from the CTS actions and SRs for the logic systems are addressed in the indicated DOCs.

A9 Deleted.

A10 The Unit 1 CTS calibration frequency for this function is not consistent with Units 2 and 3. This frequency has been reflected in the proposed BFN ISTS for Unit 1 as the same as those in the proposed ISTS for Units 2 and 3. The Unit 1 Calibration frequency for this function will be validated prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A11 CTS Table 3.2.A Reactor Low Water Level and High Drywell Pressure Functions which apply to Secondary Containment Isolation also apply to Primary Containment Isolation. These Functions specify Actions "A or (B and E)." Action B "Initiate an orderly load reduction and have Main Steam Lines isolated within eight hours" and Action E "Initiate primary containment isolation within 24 hours" are not appropriate Actions for Secondary Containment. Therefore, Actions B and E are deleted for Secondary Containment Isolation and are addressed in Primary Containment Isolation. Action A "Initiate an orderly shutdown and have the reactor in Cold Shutdown in 24 hours" is addressed in JFC L2.

A12 CTS Notes for Tables 4.2.A through 4.2.L except 4.2.D AND 4.2.K, Notes 8, 18, 19, 27 (Units 1 and 2), 28, and 29 (Unit 3), have been deleted. CTS Note 8 and Notes 27 and 28 for Units 1 and 2 and Notes 28 and 29 for Unit 3 provide a general description of instrument check, channel calibration, and channel functional test. The deletion of these notes are considered administrative since the requirements of these notes are enveloped by the proposed ISTS Definitions for CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST. CTS Notes 18 and 19 identify limitations on conditions for surveillance performance (e.g. surveillance is limited to conditions where operation of the equipment is permissible or where actuated feature surveillance are required to be met). The deletion of these notes are considered administrative since limitations on equipment operation are addressed by proposed ISTS for

the actuated features and proposed ISTS Section 3.0 (i.e., device operation resulting in an LCO not being met would result in entry into the Required Actions of the LCO per proposed ISTS LCO 3.0.2). Additionally, proposed SR 3.0.1 and 3.0.4 address that SRs shall be met during the Applicabilities of the LCOs.

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These Requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of current design basis. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 CTS 3.2.A requires primary containment isolation instrumentation to be operable when primary containment integrity is required. This includes instrumentation that initiates isolation of the reactor building and initiates the standby gas treatment system. Per CTS 3.7.A.2.a, primary containment integrity is required at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the vessel. This is more restrictive since CTS does not require the primary containment integrity when in MODE 2, not critical and < 212°F. In addition, a new Applicability has been added (proposed Notes a and b). Certain Functions will now be required to be OPERABLE during operations with a potential for draining the reactor vessel (OPDRVs), during CORE ALTERATIONS and during movement of irradiated fuel assemblies in the secondary containment. These are additional restrictions on plant operation. OPDRVs could result in a vessel draindown event and subsequent release of radioactivity, such that these instruments would be needed to isolate the Secondary Containment and start the SGT System.

M2 Proposed ISTS 3.3.6.2 Required Actions C.1.1, C.1.2, C.2.1, and C.2.2 have been added for proposed ISTS 3.3.6.2 Functions 1, 2, and 4 (refer to DOC A6 for the addition of C.1.2 and C.2.2 for proposed ISTS 3.3.6.2 Function 3). When an inoperable channel is not placed in trip within the allowed out of service time or when automatic isolation capability is not restored within the allowed out of service time, these actions require within one hour either the associated secondary containment isolation valves to be isolated or the associated secondary containment isolation valves declared inoperable and within one hour either the SGT System to be placed in operation or the SGT System declared inoperable.

These actions are comparable to CTS and proposed ISTS actions for Reactor Zone Exhaust Radiation - High Function and provide for satisfying the instrument's function or entering the Required Actions for loss of the feature. These Required Actions are consistent with the BWR Standard Technical Specifications, NUREG-1433, and are additional restrictions on plant operation.

M3 CTS 3.7.A.2.a allows "Open Vessel" physics testing at power levels not to exceed 5 MW(t). The proposed ISTS does not include provisions for this activity, therefore, this change is more restrictive. BFN is past the point in plant life where this type of testing would be performed. This change is consistent with the BWR Standard Technical Specifications, NUREG-1433.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 Information describing system design and purpose which is not needed to define the Technical Specification operability requirements has not been included in the proposed ISTS. This information is only a description of the facility and this type of information is typically provided in the FSAR. Changes to the facility and the FSAR will be controlled by the provisions of 10 CFR 50.59. Instrument identifiers (UNIDs) have not been included in the proposed ISTS, but are included in the proposed ISTS Bases as appropriate. Trip setpoints are an operational detail not directly related to the operability of the instrumentation. The Allowable Value is the required limitation of the parameter and this value is retained. The CTS "trip level settings" are equivalent to ISTS "Allowable Values". TVA's methodology for determination of setpoints utilizes the CTS "trip level settings" as the allowable value in establishing the nominal trip setpoint. The selection of nominal trip setpoints plus associated inaccuracies ensures the CTS "trip level settings" are not exceeded. TVA's setpoint methodology is consistent with RG 1.105 which endorses ISA Standard ISA-S67.04-1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants" and has been reviewed by the NRC in previous submittals, e.g. NRC letter to Mr. Oliver D. Kingsley dated January 2, 1991, Issuance of Amendment (TAC No. 77279) (TS291).

LA2 CTS Notes for Tables 4.2.A through 4.2.L except 4.2.D and 4.2.K, Note 30 (Units 1 and 3) and Note 32 (Unit 2) state that the Functional testing for the Reactor Building Ventilation Radiation Monitoring System (RBVRMS) shall consist of verifying the High Voltage Power Supply (HVPS) voltage at the Sensor and Convertors (detectors) is within its design limits and that a channel functional test as defined in Section 1.0, "Definitions" shall be performed once per 18 months as

part of the RBVRM channel calibration. The allowances and requirements of this note have been relocated to the proposed ISTS Bases for SR 3.3.6.2.2. Additionally, description of the channels, trip systems, and sensors for the RBVRMs and the requirement that both sensors for a channel must be OPERABLE (first part of CTS Table 3.2.A, Note 15 for Units 1 and 3 and Note 14 for Unit 2) are relocated to the proposed ISTS Bases. This is acceptable because the relocated information is descriptive in nature and not consistent with the type of information included in the ITS LCOs, Actions, or Surveillance Requirements.

LA3 CTS Table 3.2.A/4.2.A, SGTS Flow Functions, shut off the relative humidity heaters to protect against overheating. The Reactor Building Isolation Timer Functions are used to prevent spurious trips and system perturbations from initiating an isolation. Credit for these instruments is not assumed in any transient or accident analyses in the FSAR. As a result, the existing Technical Specification requirements for this function (including actions and surveillances) will be relocated to the Technical Requirements Manual. Any changes to these requirements will require a 10 CFR 50.59 review. Therefore, adequate assurance is provided to ensure they are adequately maintained.

LB1 The allowed out of service time (AOT) for placing a channel in trip when one trip system has inoperable, untripped channels, is extended to 12 hours for those channels common to RPS and 24 hours for all other channels. Current Technical Specifications require this action to be taken immediately. The AOT for placing a channel in trip for testing has been extended to 6 hours from 4 hours. The Channel Functional Test Frequency (STI) has been extended to once per 92 days from monthly. These AOTs and STIs have been shown to maintain an acceptable risk in accordance with previously conducted reliability analysis (NEDC 30851-P-A, Supplement 2, March 1989 and NEDC-31677-P-A, July 1990).

"Specific"

L1 Proposed Required Actions C.1.1, C.1.2, C.2.1, and C.2.2 allow 1 hour to complete required actions, whereas current Technical Specification (Note G of Table 3.2.A) requires the reactor building be isolated and the SGT system be initiated without providing a specific time period. Therefore, the proposed change is less restrictive since a finite time of one hour is allowed to complete required actions. The 1 hour Completion Time is acceptable because it allows adequate time for isolating the affected secondary containment isolation valves and placing the affected Standby Gas Treatment Subsystems(s) in operation in a safe manner.

L2 The shutdown requirements of CTS Table 3.2.A, Note 1.A have been deleted for the logic systems. An option is provided that would allow isolation of the affected lines and starting the affected SGT subsystems (Required Actions C.1.1 and C.2.1). This action conservatively compensates for the inoperable status of the instrumentation, restores the single failure capability and provides the required initiation capability of the instrumentation. If this option is not chosen, then the affected components must be declared inoperable (Required Actions C.1.2 and C.2.2), which in all likelihood would result in a unit shutdown, similar to the current action.

L3 The Action to cease refueling operations, if in progress, have been deleted. Once the affected secondary containment zone is isolated and the affected SGT System is started, the actions which would result if the instruments were actuated have already occurred. Thus, requiring further action is unnecessary. If the affected secondary containment zone is not isolated or the affected SGT System is not started, proposed Required Actions C.1.2 and C.2.2 would result in these components being declared inoperable, which would require, once the Completion Times have expired, refueling operations to cease.

L4 Note 1.H of Table 3.2.A requires that when SGTS Train or Reactor Building isolation logic trip system is found inoperable, its redundant trip systems be demonstrated operable immediately and daily thereafter (not to exceed 7 days). This requirement is deleted for several reasons. Increased testing has not been shown to demonstrate operability any better than testing at the normal SR test interval. In many cases, increased testing adds to the failure rates of components by increasing wear and tear. Common mode failure analysis in conjunction with loss of function analyses provide adequate assurance of redundant system operability. Loss of function determination program controls are provided by BFN ISTS 5.5.11.

L5 For the affected isolation functions, CTS Table 3.2.A, Note 1, requires the appropriate action listed in Table 3.2.A be taken if, for the same function, the first column of Table 3.2.A cannot be met for all trip systems. Proposed BFN ISTS 3.3.6.2, Required Action B.1, provides a one hour Completion Time from discovery of loss of isolation capability to make this declaration. Proposed Required Action B.1 does not require further action unless isolation capability is lost. This is acceptable since the Function is still capable of performing its design function. The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in isolation capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels.

L6 CTS Table 4.2.A requires a functional test of the Reactor Building Isolation (refueling floor), Reactor Building Isolation (reactor zone), SGTS Train A, SGTS Train B, and SGTS Train C Logic every six months. The table requires a functional test of the Group 6 Logic every 18 months (once per operating cycle). The Group 6 Logic provides input to the Reactor Building Isolation (refueling floor), Reactor Building Isolation (reactor zone), SGTS Train A, SGTS Train B, and SGTS Train C Logic. The proposed Logic System Functional Test (SR 3.3.6.2.4) which based on definition includes "all required logic components (i.e., all required relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device" would include the Group 6 Logic components. To be consistent a frequency of 18 months was specified for SR 3.3.6.2.4.

Previously the functional test of the Reactor Building Isolation (refueling floor), Reactor Building Isolation (reactor zone), SGTS Train A, SGTS Train B, and SGTS Train C Logic included the initiation of the actuated device. These devices will be actuated more frequently than once per six months based on proposed SR 3.6.4.2.1 which requires each automatic SCIV be actuated every 92 days and SR 3.6.4.3.1 which requires each SGT subsystem be operated at least 10 continuous hours every 31 days.

Only four test deficiencies involving required relays were noted in a review of performance history since the beginning of 1991 for Surveillance 0-SI-4.2.A-12, Standby Gas Treatment Blower and Heater Logic Functional Test; 0-SI-4.2.A-17, Refueling Floor Isolation Logic System Functional Test; and 1, 2, 3-SI-4.2.A-20, Reactor Zone Isolation Logic System Functional Test. None of the four deficiencies affected acceptance criteria steps in the surveillances.

Based on the above information, it is acceptable to extend the test frequency from six months to 18 months to be consistent with other Group 6 Logic System Functional Test frequencies and to allow performance of the surveillance under the conditions that apply during a plant outage which will reduce the potential for an unplanned transient with the reactor at power.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.7.1 - CONTROL ROOM EMERGENCY
VENTILATION SYSTEM INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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. In the specific case of the CREVS instrumentation, these functions have been combined into one Specification.

Editorial rewording (either adding or deleting) is done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 CTS Table 3.2.G contains Note 3 which references CTS Table 3.2.A for instrumentation that isolates primary containment as applying to control room isolation instrumentation. (CTS Table 3.2.A also applies to secondary containment isolation) The Functions in CTS applicable to control room isolation instrumentation specify Actions "A or (B and E)", "F", and "G." These Actions require:

A - "initiate an orderly shutdown and have the reactor in COLD shutdown in 24 hours."
B - "initiate an orderly load reduction and have Main Steam Lines isolated within eight hours."
E - "initiate primary containment isolation within 24 hours."
F - "the handling of spent fuel will be prohibited and all operations over spent fuels and open reactor wells shall be prohibited."
G - "isolate the Reactor Building and Start Standby Gas Treatment System."

These Actions are not appropriate for Control Room Emergency Ventilation System. Therefore, Actions A, B, E, F, and G are not incorporated in ITS for Control Room Emergency Ventilation System Instrumentation and are addressed in Primary Containment and Secondary Containment Isolation.

A3 The Frequency "once each operating cycle" has been changed to "18 months." The Frequency of "once per six months" has been changed to "184 days." The Frequency of "once per 3 months" has been changed to "92 days." The Frequency of "once per day" has been changed to "24 hours." These are changes in nomenclature and are considered administrative.

A4 Not used.

A5 The Unit 1 CTS calibration frequency for this function is not consistent with Units 2 and 3. This frequency has been reflected in the proposed BFN ISTS for Unit 1 as the same as those in the proposed ISTS for Units 2 and 3. The Unit 1 Calibration frequency for this function will be validated prior to Unit 1 recovery and changes to the proposed BFN ISTS for Unit 1 will be made as necessary.

A6 CTS Table 3.2.G identifies the minimum number of OPERABLE instrument channels on a total bases, and thus, indicates two channels of Control Room Air Supply Duct Radiation-High Function are required to be OPERABLE. The proposed ISTS Table 3.3.7.1-1 identifies the minimum number of operable instrument channels on a per trip system bases, and thus, indicates one channel per trip system of Control Room Air Supply Duct Radiation - High Function is required to be OPERABLE. Since there are two trip systems for Control Room air Supply Duct Radiation - High Radiation - High Function, the total number of channels required to be maintained OPERABLE are the same in both the CTS and ISTS, and thus, this change is considered administrative.

A7 CTS NOTES FOR TABLES 4.2.A THROUGH 4.2.L EXCEPT 4.2.D AND 4.2.K, Notes 8, 18, 27 (Units 1 and 2), 28, and 29 (Unit 3), have been deleted. CTS Note 8 and Notes 27 and 28 for Units 1 and 2 and Notes 28 and 29 for Unit 3 provide a general description of instrument check, channel calibration, and channel functional test. The deletion of these notes are considered administrative since the requirements of these notes are enveloped by the proposed ISTS Definitions for CHANNEL CHECK, CHANNEL CALIBRATION, and CHANNEL FUNCTIONAL TEST. CTS Note 18 identifies limitations on conditions for surveillance performance (e.g., surveillance is limited to conditions where operation of the equipment is permissible). The deletion of this note is considered administrative since limitations on equipment operation are addressed by proposed ISTS for the actuated features and proposed ISTS Section 3.0 (i.e., device operation resulting in an LCO not being met would result in entry into the Required Actions of the LCO per proposed ISTS LCO 3.0.2).

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These Requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 Proposed Required Actions B.1 and C.1 have been added requiring that the CREV System be declared inoperable upon discovery of loss of CREV initiation capability by proposed Functions 1, 2, 3, and 4. This is in addition to CTS requirements specified for these functions in Table 3.2.A for primary containment and reactor building isolation instrumentation. Proposed Required Action E.2 has been added to require the CREV System be declared inoperable when the required action and associated Completion Time of Condition B, C or D are not met. For Conditions B and C this is in addition to CTS requirements specified for these functions in Table 3.2.A. For Condition D, this requirement already existed as part of Note (2) to Table 3.2.G. Proposed Required Action E.1 allows the option of placing CREVS in the pressurization mode of operation. Since ACTION E is being added for Functions 1, 2, 3, and 4, the option allowed by proposed Required Action E.1 is considered more restrictive.

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 Information describing system design and purpose which is not needed to define the Technical Specification operability requirements has not been included in the proposed ISTS. This information is only a description of the facility and this type of information is typically provided in the FSAR. Changes to the facility and the FSAR will be controlled by the provisions of 10 CFR 50.59. Instrument identifiers (UNIDs) have not been included in the proposed ISTS, but are included in the proposed ISTS Bases as appropriate. Trip setpoints are an operational detail not directly related to the operability of the instrumentation. The Allowable Value is the required limitation of the parameter and this value is retained. The CTS "trip level settings" are equivalent to ISTS "Allowable Values". TVA's methodology for determination of setpoints utilizes the CTS "trip level settings" as the allowable value in

establishing the nominal trip setpoint. The selection of nominal trip setpoints plus associated inaccuracies ensures the CTS "trip level settings" are not exceeded. TVA's setpoint methodology is consistent with RG 1.105 which endorses ISA Standard ISA-S67.04-1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants" and has been reviewed by the NRC in previous submittals, e.g., NRC letter to Mr. Oliver D. Kingsley dated January 2, 1991, Issuance of Amendment (TAC No. 77279) (TS291).

LA2 CTS NOTES FOR TABLES 4.2.A THROUGH 4.2.L EXCEPT 4.2.D AND 4.2.K, Note 30 (Units 1 and 3) and Note 32 (Unit 2) state that the Functional testing for the Reactor Building Ventilation Radiation Monitoring System (RBVRMS) shall consist of verifying the High Voltage Power Supply (HVPS) voltage at the Sensor and Convertors (detectors) is within it design limits and that a channel functional test as defined in Section 1.0, "Definitions" shall be performed once per 18 months as part of the RBVRM channel calibration. The allowances and requirements of this note have been relocated to the proposed ISTS Bases for SR 3.3.7.1.2. Additionally, description of the channels, trip systems, and sensors for the RBVRMs and the requirement that both sensors for a channel must be OPERABLE (first part of CTS Table 3.2.A, Note 15 for Units 1 and 3 and Note 14 for Unit 2) are relocated to the proposed ISTS 3.3.6.2 Bases (proposed Bases for ISTS 3.3.7.1 (BACKGROUND) references 3.3.6.2 Bases for description of channels and trip systems since they are common).

LB1 The allowed out of service time (AOT) for placing a channel in trip when one trip system has inoperable, untripped channels, is extended to 12 hours for those channels common to RPS and 24 hours for all other channels. The AOTs for placing a channel in trip for required surveillance testing has been extended from 4 hours to 6 hours. The channel functional test frequency (STI) has been extended to once per 92 days from monthly. These AOTs and STIs have been shown to maintain a acceptable risk in accordance with previously conducted reliability analysis (NEDC 30851-P-A, Supplement 2, March 1989, NEDC-31677-P-A, July 1990, and GENE-770-06-1, February 1991).

"Specific"

L1 An additional Required Action has been added (Required Action E.1) to place the associated CREV subsystems in the pressurization mode, if the channels are not restored to OPERABLE status or tripped within the allowed time. Since CTS Table 3.2.G Note (2) currently requires the CREV system to be declared inoperable if one Control Room air supply duct radiation monitor is not operable within 30 days, the option of placing the associated CREV subsystem in operation is less restrictive

for proposed ISTS 3.3.7.1 Functions 5. Initiation of the associated subsystems returns the CREV System to a status where the instrumentation has performed its function (initiation of the system is the function of the instruments); thus, a shutdown is not necessary.

JUSTIFICATION FOR CHANGES
BFN ISTS 3.3.8.1 - LOSS OF POWER (LOP) INSTRUMENTATION

ADMINISTRATIVE

A1 Reformatting and renumbering are in accordance with the BWR Standard Technical Specifications, NUREG 1433. 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 done to make consistent with NUREG-1433. During ISTS 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 the BWR Standard Technical Specifications, NUREG-1433. Since the design is already approved, adding more detail does not result in a technical change.

A2 This change proposes to add a Note which will allow separate Condition entry for each channel. This change provides more explicit instructions for proper applications of the Actions for Technical Specifications compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," the Note ("Separate Condition entry ...") and "in one or more Functions" provides more explicit direction of the current interpretation of the existing Specifications. This change is considered administrative and is consistent with NUREG-1433.

TECHNICAL CHANGE - MORE RESTRICTIVE

Those items that are identified as More Restrictive (M type JFCs) contain requirements that are more restrictive than the Current Technical Specifications. These Requirements are based on the Standard Technical Specifications for BWR/4, NUREG-1433, modified to reflect BFN specific design. These additional requirements have been determined to be appropriate for BFN based on a review of the current design bases. Adoption of these More Restrictive requirements will provide additional assurance of conditions which will protect the health and safety of plant personnel and the public. Based on this, it is concluded that these enhancements to the Technical Specifications should be included in the ITS.

M1 An additional Applicability has been added, requiring the instruments to be OPERABLE when the associated diesel generators (DGs) are required to be OPERABLE by LCO 3.8.2, AC Sources-Shutdown. This essentially adds a MODE 4 and 5 applicability when the DGs are required in these MODES. This is consistent with the BWR Standard Technical Specifications, NUREG 1433 and is an additional restriction on plant operation.

M2 Current TS 4.9.A.4.c requires degraded voltage relays to be calibrated annually. BFN has been performing this calibration every 184 days due to recent operating performance of these relays. As such, the TS calibration interval has been reduced to 184 days (proposed SR 3.3.8.1.1).

TECHNICAL CHANGE - LESS RESTRICTIVE

"Generic"

LA1 System design and operational details have been relocated to the Bases and procedures. Trip setpoints and the reset functions are operational details that are not directly related to the operability of the required instrumentation. The Allowable Value is the required limitation for the parameter and this value is retained in SR 3.3.8.1.2. The CTS "trip level settings" are equivalent to ITS "allowable Values." TVA's methodology for determination of setpoints utilizes the CTS "trip level settings" as the allowable value in establishing the nominal trip setpoint. The selection of nominal trip setpoints plus associated inaccuracies ensures the CTS "trip level settings" are not exceeded. TVA's setpoint methodology is consistent with RG 1.105 which endorses ISA Standard ISA-S67.04-1982 "Setpoints for Nuclear Safety Related Instrumentation Used in Nuclear Power Plants" and has been reviewed by the NRC in previous submittals, e.g. NRC letter to Mr. Oliver D. Kingsley dated January 2, 1991, Issuance of Amendment (TAC No. 77279)(TS291). Details relating to system design and operation (e.g., description of action of instrumentation) are also unnecessary in

the LCO and have been relocated to the Bases and procedures. The design features and system operation are also described in the FSAR. In addition, requirements to record values during Surveillance Requirements have been relocated to plant procedures. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

LA2 Details of the methods for performing surveillances are relocated to the Bases and procedures. The design features and system operation which dictate the methods are described in the FSAR. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Process in Chapter 5 of the Technical Specifications. Changes to the FSAR and procedures will be controlled by the provisions of 10 CFR 50.59.

"Specific"

L1 A new ACTION has been added (proposed ACTION E) to require declaring the DG inoperable (and taking the appropriate actions in the associated DG Specification) if a channel is not restored when required. Current Technical Specifications 3.9.B.15 (Units 1 and 2) and 3.9.B.13 (Unit 3) require an orderly shutdown be initiated and the reactor to be in Cold Shutdown within 24 hours. Since these instruments provide start signals to the DGs (i.e., it supports DG OPERABILITY), it is more appropriate to declare the DG inoperable (proposed ACTION E). The current requirements are overly restrictive, in that if the diesel were inoperable for other reasons, a 7 day restoration time is provided; yet currently if the ACTION for the instruments can not be met but the diesel is otherwise fully OPERABLE, a shutdown is required.

L2 Deleted (NRC Question 3.3.8.1-2).

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS MARKUP

Replaced page 88 of 478 (STS page 3.3-4) with page 88 of 478 Rev. 2
Replaced page 91 of 478 Rev. 1 (STS page 3.3-7) with page 91 of 478 Rev. 2
Replaced page 92 of 478 Rev. 1 (STS page 3.3-8) with page 92 of 478 Rev. 2
Replaced page 104 of 478 (STS page 3.3-20) with page 104 of 478 Rev. 2
Replaced page 105 of 478 Rev. 1 (STS page 3.3-21) with page 105 of 478 Rev. 2
Replaced page 109 of 478 Rev. 1 (STS page 3.3-25) with page 109 of 478 Rev. 2
Replaced page 112 of 478 Rev. 1 (STS page 3.3-28) with page 112 of 478 Rev. 2
Replaced page 116 of 478 (STS page 3.3-32) with page 116 of 478 Rev. 2
Replaced page 117 of 478 Rev. 1 (STS page 3.3-33) with page 117 of 478 Rev. 2
Replaced page 118 of 478 (STS page 3.3-34) with page 118 of 478 Rev. 2
Replaced page 120 of 478 Rev. 1 (STS page 3.3-36) with page 120 of 478 Rev. 2
Replaced page 121 of 478 Rev. 1 (STS page 3.3-37) with page 121 of 478 Rev. 2
Replaced page 122 of 478 Rev. 1 (STS page 3.3-38) with page 122 of 478 Rev. 2
Replaced page 138 of 478 Rev. 1 (STS page 3.3-52) with page 138 of 478 Rev. 2
Replaced page 139 of 478 Rev. 1 (STS page 3.3-53) with page 139 of 478 Rev. 2
Replaced page 140 of 478 Rev. 1 (STS page 3.3-54) with page 140 of 478 Rev. 2
Replaced page 144 of 478 (STS page 3.3-58) with page 144 of 478 Rev. 2
Replaced page 146 of 478 Rev. 1 (STS page 3.3-60) with page 146 of 478 Rev. 2
Replaced page 149 of 478 Rev. 1 (STS page 3.3-62) with page 149 of 478 Rev. 2
Replaced page 152 of 478 (insert page 3.3-64A) with page 152 of 478 Rev. 2
Replaced page 163 of 478 (insert page 3.3-73A) with page 163 of 478 Rev. 2
Replaced page 167 of 478 Rev. 1 (insert page 3.3-75A) with page 167 of 478 Rev. 2
Replaced page 167a of 478 Rev. 1 (insert page 3.3-75B) with page 167a of 478 Rev. 2
Replaced page 169 of 478 (STS Table 3.3.8.1.1 page 1 of 1) with page 169 of 478 Rev. 2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.~~8~~ 4 (P1)	Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.~~9~~ 5 (P1)	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to withdrawing SRMs from the fully inserted position
SR 3.3.1.1.~~7~~ 6 (P1)	----------NOTE---------- Only required to be met during entry into MODE 2 from MODE 1. ---------- Verify the IRM and APRM channels overlap.	7 days
SR 3.3.1.1.~~8~~ 7 (P1)	Calibrate the local power range monitors.	1000 ~~MWD/T average core exposure effective full power hours~~ (P6) stet
SR 3.3.1.1.~~9~~ 8 (P1)	Perform CHANNEL FUNCTIONAL TEST.	~~192~~ days (B2)
~~SR 3.3.1.1.10~~	~~Calibrate the trip units.~~	~~[92] days~~ (B3) (P2)
SR 3.3.1.1.9 (P91)	---------- NOTES ---------- 1. Neutron detectors are excluded. 2. For Functions 1 and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. ---------- Perform CHANNEL CALIBRATION.	92 days

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	(B2) This column ALLOWABLE VALUE
1. Intermediate Range Monitors	[2 (P5) move down	~~3~~	G	SR 3.3.1.1.1 SR 3.3.1.1.~~3~~ 3 SR 3.3.1.1.~~5~~ 5 SR 3.3.1.1.~~6~~ 6 SR 3.3.1.1.~~9~~ 9 SR 3.3.1.1.~~14~~ 14	≤ ~~120/125~~ divisions of full scale
a. Neutron Flux –High	5(a)	~~3~~	H	SR 3.3.1.1.1 SR 3.3.1.1.~~4~~ 4 SR 3.3.1.1.~~9~~ 9 SR 3.3.1.1.~~14~~ 14	≤ ~~120/125~~ divisions of full scale
b. Inop	2	~~3~~	G	SR 3.3.1.1.~~3~~ 3 SR 3.3.1.1.~~14~~ 14	NA
	5(a)	~~3~~	H	SR 3.3.1.1.~~4~~ 4 SR 3.3.2.2.~~15~~ 14	NA
2. Average Power Range Monitors					
a. Neutron Flux –High, Setdown	2	~~2~~	G	SR 3.3.1.1.1 SR 3.3.1.1.~~3~~ 3 SR 3.3.1.1.~~6~~ 6 SR 3.3.1.1.~~7~~ 7 SR 3.3.1.1.~~9~~ 9 SR 3.3.1.1.~~14~~ 14	≤ ~~20~~ 15% RTP
ADD INSERT 11					
b. Flow Biased Simulated Thermal Power –High	1	~~2~~ (B2) This column	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.~~7~~ 7 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14 ~~SR 3.3.1.1.15~~ ~~SR 3.3.1.1.17~~	≤ ~~0.58~~ W + ~~62%~~ RTP and ≤ ~~(115.51%~~ 120 % RTP~~(b)~~ (P6)
		(P1) and (P4) This column			(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

~~(b) [0.58 W + 62% - 0.58 ΔW]RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."~~

(P6) (P92)

5(a)	2	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.9 SR 3.3.1.1.14	

INSERT 11

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Average Power Range Monitors (continued)					
c. Fixed Neutron Flux – High	1	~~2~~	F	SR 3.3.1.1.1 SR 3.3.1.1.2 SR 3.3.1.1.~~8~~ 7 SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.~~11~~ 9 SR 3.3.1.1.~~13~~ 14 P4 ~~SR 3.3.1.1.17~~	≤ ~~120~~% RTP
B2 d. Downscale	1	~~2~~	F	SR 3.3.1.1.~~8~~ 7 SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.~~13~~ 14	≥ ~~3~~% RTP
e. Inop	1,2	~~2~~	G	SR 3.3.1.1.~~8~~ 7 SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.~~13~~ 14	NA
ADD INSERT A					
3. Reactor Vessel Steam Dome Pressure – High	1,2	~~2~~	G	SR 3.3.1.1.1 SR 3.3.1.1.~~9~~ 8 P2 ~~[SR 3.3.1.1.10]~~ SR 3.3.1.1.~~12~~ 10 SR 3.3.1.1.~~13~~ 14 P4 ~~SR 3.3.1.1.17~~	≤ ~~[1054]~~ 1055 psig
4. Reactor Vessel Water Level – Low, Level 3	1,2	~~2~~	G	SR 3.3.1.1.1 SR 3.3.1.1.~~9~~ 8 P2 ~~[SR 3.3.1.1.10]~~ SR 3.3.1.1.13 SR 3.3.1.1.~~13~~ 14 P4 ~~SR 3.3.1.1.17~~	≥ ~~[10]~~ 538 inches above vessel zero
5. Main Steam Isolation Valve – Closure	1	~~8~~	F	SR 3.3.1.1.~~9~~ 8 SR 3.3.1.1.13 SR 3.3.1.1.~~13~~ 14 ~~SR 3.3.1.1.17~~	≤ ~~10~~% closed
6. Drywell Pressure – High	1,2	~~2~~	G	P59 ~~SR 3.3.1.1.1~~ SR 3.3.1.1.~~9~~ 8 P2 ~~[SR 3.3.1.1.10]~~ SR 3.3.1.1.13 SR 3.3.1.1.~~13~~ 14	≤ ~~[1.92]~~ 2.5 psig

B2 This column

B2 This column

(continued)

P1 This column unless indicated otherwise

P92

5(a)	2	H	SR 3.3.1.1.7 SR 3.3.1.1.8 SR 3.3.1.1.14	

INSERT A

REV. 2

Table 3.3.2.1-1 (page 1 of 1)
Control Rod Block Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Rod Block Monitor				
a. Low Power Range -Upscale (Flow Biased)	(a), (b)	[2]	SR 3.3.2.1.1 ~~SR 3.3.2.1.4~~ SR 3.3.2.1.7	~~≤ [115.5/125] divisions of full scale~~ (c)
b. Intermediate Power Range -Upscale	(b)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≤ [109.7/125] divisions of full scale
c. High Power Range -Upscale	(c)/(d)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.4 SR 3.3.2.1.7	≤ [105.9/125] divisions of full scale
b. Inop	a (f), b (f)	[2]	SR 3.3.2.1.1	NA
c. Downscale	a (f), b (f)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.7	≥ 3% RTP ~~≥ [93/125] divisions of full scale~~
f. Bypass Time Delay	(d)/(e)	[2]	SR 3.3.2.1.1 SR 3.3.2.1.7	≤ [2.0] seconds
2. Rod Worth Minimizer	1 (f), 2 (f)	[1]	SR 3.3.2.1.2 SR 3.3.2.1.3 SR 3.3.2.1.5 SR 3.3.2.1.8	NA
3. Reactor Mode Switch -Shutdown Position	d (f)	[2]	SR 3.3.2.1.6	NA

(a) THERMAL POWER ≥ [29]% and ≤ [64]% RTP and MCPR < 1.70.
(b) THERMAL POWER > [64]% and ≤ [84]% RTP and MCPR < 1.70.
(c) THERMAL POWER > [84]% and < 90% RTP and MCPR < 1.70.

a (f) THERMAL POWER ≥ 90% RTP and MCPR < ~~1.70~~ 1.44
b (f) THERMAL POWER ≥ ~~[64]%~~ 29% and < 90% RTP and MCPR < ~~1.70~~ 1.75
c (f) With THERMAL POWER ≤ [10]% RTP.
d (f) Reactor mode switch in the shutdown position.
(e) Less than or equal to the Allowable Value specified in the COLR.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 ~~[Three]~~ Two channels of feedwater and main turbine high water level trip instrumentation per trip system shall be OPERABLE. (B2) (P11)

APPLICABILITY: THERMAL POWER ≥ ~~[25]~~ 15% RTP. (B2)

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more (P11) feedwater and main turbine high water level trip channel(s) (P11) inoperable, in one trip system.	A.1	Place channel in trip.	7 days
B. ~~Two~~ One (P11) or more feedwater and main turbine high water level trip channels inoperable in each trip system. (P11)	B.1	Restore feedwater and main turbine high water level trip capability.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Reduce THERMAL POWER to < ~~[25]~~ 15% RTP. (B2)	4 hours

SURVEILLANCE REQUIREMENTS

PS7

NOTE
These SRs apply to each Function in Table 3.3.3.1-1.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1.1	Perform CHANNEL CHECK for each required PAM instrumentation channel	31 days
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION for each required PAM instrumentation channel except for the Drywell and Torus H_2 Analyzer Functions (AND THE Reactor Pressures)	[18] months
SR 3.3.3.1.2	Perform CHANNEL CALIBRATION of the Drywell + Torus H_2 Analyzer Functions.	92 days
SR 3.3.3.1.3	Perform CHANNEL CALIBRATION of the Reactor Pressure ~~indications~~ Functions	184 days

PS7 PS7 PS6 PS6 F F

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.3.2.~~2~~ 1 P1	Verify each required control circuit and transfer switch is capable of performing the intended function.	~~[18]~~ months B2
SR 3.3.3.~~2.3~~ 3~~4~~ P1	Perform CHANNEL CALIBRATION for each required instrumentation channel except for the Suppression Pool Water Level Function.	~~[18]~~ months
SR 3.3.3.2.~~3~~2 P1 P66	Perform CHANNEL CALIBRATION for the Suppression Pool Water Level Function.	184 days (Units 1 and 2) 18 Months (Unit 3)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
B3 SR 3.3.4.1.2	Calibrate the trip units.	[92] days
SR 3.3.4.1.3	Perform CHANNEL CALIBRATION. The Allowable Values shall be: TSV-Closure: ≥ [10]% closed; and (B2) (P18) TCV Fast Closure, Trip Oil Pressure-Low: ≥ ~~[600]~~ 550 psig. (B2)	[18] months (B2)
SR 3.3.4.1.4	Perform LOGIC SYSTEM FUNCTIONAL TEST including breaker actuation.	[18] months (B2)
SR 3.3.4.1.~~8~~ 2 (P1)	Verify TSV-Closure and TCV Fast Closure, Trip Oil Pressure-Low Functions are not bypassed when THERMAL POWER is ≥ [30]% RTP. (B2)	[18] months (B2)
SR 3.3.4.1.6	----------NOTE---------- Breaker [interruption] time may be assumed from the most recent performance of SR 3.3.4.1.7. ---------- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	[18] months on a STAGGERED TEST BASIS
SR 3.3.4.1.7	Determine RPT breaker [interruption] time.	60 months

(P4)

3.3 INSTRUMENTATION

3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

LCO 3.3.4.2 Two channels per trip system for each ATWS-RPT instrumentation Function listed below shall be OPERABLE:

a. Reactor Vessel Water Level – Low ~~Low, Level 2~~ Low, Level 2; and

b. Reactor Steam Dome Pressure – High.

APPLICABILITY: MODE 1.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more channels inoperable.	A.1 Restore channel to OPERABLE status.	14 days
	OR	
	A.2 --------NOTE--------- Not applicable if inoperable channel is the result of an inoperable breaker. ---------- Place channel in trip.	14 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One Function with ATWS-RPT trip capability not maintained.	B.1 Restore ATWS-RPT trip capability.	72 hours
C. Both Functions with ATWS-RPT trip capability not maintained.	C.1 Restore ATWS-RPT trip capability for one Function.	1 hour
D. Required Action and associated Completion Time not met.	~~D.1 Remove the associated recirculation pump from service.~~	~~6 hours~~
	OR	
	D.2 (D.1) Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

----------NOTE----------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains ATWS-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.2.1	Perform CHANNEL CHECK of the Reactor Vessel water Level-Low, Function (Low, Level 2)	~~12~~ 24 hours

(continued)

3.3 INSTRUMENTATION

3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

LCO 3.3.5.1 The ECCS instrumentation for each Function in Table 3.3.5.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5.1-1.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more channels inoperable.	A.1	Enter the Condition referenced in Table 3.3.5.1-1 for the channel.	Immediately
B. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	B.1	--------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.a, 1.b, 2.a, ~~and~~ 2.b, and 2.c ----------	
		Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of ~~feature~~ Function(s) initiation capability ~~for feature(s)~~ for features in both ~~divisions~~ trip systems divisions
	AND		
		PAGE 120 OF 478	(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 --------NOTE--------- Only applicable for Functions 3.a and 3.b. ----------	
	Declare High Pressure Coolant Injection (HPCI) System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND	
	B.3 Place channel in trip.	24 hours
C. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	C.1 --------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Functions 1.c ~~1.e~~, ~~2.c~~ 2.c, 2.d, and 2.f. ----------	
	Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of ~~subsystem~~ initiation capability ~~for feature(s)~~ in both ~~divisions~~ ~~subsystems~~ ~~HPCS System~~
	AND	
	C.2 Restore channel to OPERABLE status.	24 hours

P11 1.e P99 P11 ~~Function(s)~~ for features divisions

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	D.1 --------NOTE--------- Only applicable if HPCI pump suction is not aligned to the suppression pool. ---------- Declare HPCI System inoperable.	1 hour from discovery of loss of HPCI initiation capability
	AND	
	D.2.1 Place channel in trip.	24 hours
	OR	
	D.2.2 Align the HPCI pump suction to the suppression pool.	24 hours
E. As required by Required Action A.1 and referenced in Table 3.3.5.1-1.	E.1 --------NOTES-------- 1. Only applicable in MODES 1, 2, and 3. 2. Only applicable for Function~~s~~ 1.d ~~and 2.g.~~ ---------- Declare supported ECCS feature(s) inoperable when its redundant feature ECCS initiation capability is inoperable.	1 hour from discovery of loss of ~~subsystem~~ Function initiation capability ~~for subsystems in both divisions~~ ~~in both subsystems~~ for subsystems in both divisions
	AND	

P81 P11 P94 P99 P91

(continued)

3.3 INSTRUMENTATION

3.3.6.1 Primary Containment Isolation Instrumentation

LCO 3.3.6.1 The primary containment isolation instrumentation for each Function in Table 3.3.6.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.6.1-1.

P64

NOTE
ONLY APPLICABLE FOR FUNCTION 1.d IF TWO OR MORE CHANNELS ARE INOPERABLE

~~NOTE~~
~~Not applicable for Function 1.d when 15 of 16 channels are OPERABLE~~

ACTIONS

NOTE
Separate Condition entry is allowed for each channel

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Place channel in trip.	12 hours for Functions 2.a, 2.b, ~~and 6.b~~ P83 P23 5.h, 6.b, and 6.c AND 24 hours for Functions other than Functions 2.a, 2.b, ~~and 6.b~~ P83 P64
	AND NOTE Only applicable for Function 1.d when 15 of 16 channels are OPERABLE P64 27 A.2 Place channel in trip	30 days
B. One or more ~~automatic~~ P11 Functions with isolation capability not maintained.	B.1 Restore isolation capability.	1 hour OR 4 hours for Function 1.d when normal ventilation is not available

P25

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Enter the Condition referenced in Table 3.3.6.1-1 for the channel.	Immediately
D. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	D.1	Isolate associated main steam line (MSL) ~~the affected penetration flow path(s)~~	12 hours ASSOCIATED MAIN STEAM LINE (MSL)
	OR		
	D.2.1	Be in MODE 3. P32	12 hours
	AND		
	D.2.2	Be in MODE 4.	36 hours
E. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	E.1	Be in MODE 2.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	F.1	Isolate the affected penetration flow path(s).	1 hour
P35 G. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	G.1	Isolate the affected penetration flow path(s).	24 hours

(continued)

~~G.1 NOTE
Only applicable for Functions 2.a and 2.b channels which are inoperable as a result of inoperable actuation logic.
Isolate the affected penetration flow path(s).
1 hour~~ P94

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~G.~~ H. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. P1 ~~OR~~	~~G.1~~ H.1 Be in MODE 3. P1 ~~G.2~~ G.1	12 hours
	AND	
	~~G.~~ H.2 Be in MODE 4. P1 G.2	36 hours
OR Required Action and associated Completion Time for Condition F or G not met. P35		
~~H.~~ I. As required by Required Action C.1 and referenced in Table 3.3.6.1-1. P1	~~H.~~ I.1 Declare associated standby liquid control subsystem (SLC) inoperable. P1 P9	1 hour
	OR	
	~~H.~~ I.2 Isolate the Reactor Water Cleanup System. P1	1 hour
~~I.~~ J. As required by Required Action C.1 and referenced in Table 3.3.6.1-1.	~~J.~~ I.1 Initiate action to restore channel to OPERABLE status. P1	Immediately
	OR	
	~~J.~~ I.2 Initiate action to isolate the Residual Heat Removal (RHR) Shutdown Cooling System. P1	Immediately

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Rev 2

Table 3.3.6.1-1 (page 2 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment Isolation		B2 ~~[2]~~	G		
a. Reactor Vessel Water Level –Low, Level 3	1,2,3	~~[2]~~	~~H~~	SR 3.3.6.1.1 SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~6~~ 5 SR 3.3.6.1.~~7~~ 6 P4 ~~SR 3.3.6.1.8~~	≥ ~~[10]~~ 538 inches above vessel zero (B2)
b. Drywell Pressure –High	1,2,3	~~[2]~~	~~H~~ G	P59 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~6~~ 5 SR 3.3.6.1.~~7~~ 6 P4 ~~SR 3.3.6.1.8~~	≤ ~~[1.92]~~ 2.5 psig (P5) (B2) (P1)
P39 c. Drywell Radiation –High	1,2,3	~~[1]~~	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ [138] R/hr
P87 d. Reactor Building Exhaust Radiation –High	1,2,3	[2]	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ [60] mR/hr
e. Refueling Floor Exhaust Radiation –High	1,2,3	[2]	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.6 SR 3.3.6.1.7 SR 3.3.6.1.8	≤ [20] mR/hr
P35 f. Manual Initiation	1,2,3	[1 per group]	G	SR 3.3.6.1.7	NA
3. High Pressure Coolant Injection (HPCI) System Isolation					
a. HPCI Steam Line Flow –High	1,2,3	~~[1]~~	F	P74 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~6~~ 5 SR 3.3.6.1.~~7~~ 6 P4 ~~SR 3.3.6.1.8~~	≤ ~~[303]~~ 150 % rated steam flow (90 psi) (B2) (P1)

(continued)

Table 3.3.6.1-1 (page 4 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. Reactor Core Isolation Cooling (RCIC) System Isolation		B2			450" H_2O B2
a. RCIC Steam Line Flow – High	1,2,3	~~[1]~~	F	P74 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~6~~ 5 P1 SR 3.3.6.1.~~7~~ 6 P4 ~~SR 3.3.6.1.8~~	≤ ~~[307%]~~ 150 rated steam flow
b. RCIC Steam Supply Line Pressure – Low	1,2,3	~~[2]~~ 3	F	P74 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~8~~ 5 P1 SR 3.3.6.1.~~7~~ 6	≥ ~~[60]~~ 50 psig P2 P5
c. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	~~[2]~~ 3	F	P74 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~6~~ 5 P1 SR 3.3.6.1.~~7~~ 6	≤ ~~[20]~~ psig B2
P40 d. Drywell Pressure – High	1,2,3	[1]	F	SR 3.3.6.1.1 SR 3.3.6.1.2 [SR 3.3.6.1.3] SR 3.3.6.1.6 SR 3.3.6.1.7 [SR 3.3.6.1.8]	≤ [1.92] psig
INSERT From Next page					
P11 e. RCIC ~~Suppression Pool Ambient~~ Steam Line Space Torus Area Temperature – High (Exit) P11	1,2,3	~~[1]~~ 2	F	P74 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~6~~ 3 P43 P1 SR 3.3.6.1.~~7~~ 6	≤ ~~[169]~~ 155 °F B2
INSERT 3.3-60A					
~~f.~~ Suppression Pool Area Temperature – Time Delay Relays	1,2,3	[1]	F	SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ [NA] [minutes]
~~g.~~ RCIC ~~Suppression Pool~~ Area ~~Differential~~ Temperature – High	1,2,3	~~[1]~~	F	SR 3.3.6.1.1 SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.6 SR 3.3.6.1.7	≤ ~~[42]~~ °F
h. Emergency Area Cooler Temperature – High	1,2,3	[1]	F	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.6 SR 3.3.6.1.7	≤ [169] °F
P40					

(continued)

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
6. Shutdown Cooling System Isolation		B2			
a. Reactor Steam Dome Pressure – High	1,2,3	~~1~~ 2	F	P74 ~~SR 3.3.6.1.1~~ SR 3.3.6.1.2 P2 ~~[SR 3.3.6.1.3]~~ SR 3.3.6.1.~~4~~ 5 SR 3.3.6.1.~~7~~ 6	≤ ~~[145]~~ 115 psig (B2)
b. Reactor Vessel Water Level – Low, Level 3	3,4,5	~~2~~ 2 (B2)	P1 b I	SR 3.3.6.1.1 SR 3.3.6.1.2 ~~[SR 3.3.6.1.3]~~ P2 SR 3.3.6.1.~~4~~ 5 SR 3.3.6.1.~~7~~ 6 P1	≥ ~~[100]~~ 538 inches (B2) above vessel zero
c. Drywell Pressure – High	1,2,3	2	F	SR 3.3.6.1.2 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 2.5 psig

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~~(c) Only one trip system required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.~~

~~(b) Only one channel per trip system for one RHR Shutdown Cooling (SDC) Supply isolation valve required in MODES 4 and 5 when RHR Shutdown Cooling System integrity maintained.~~

P1 P95

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ONLY ONE CHANNEL PER TRIP SYSTEM REQUIRED IN MODES 4 AND 5 WHEN RHR SHUTDOWN COOLING SYSTEM INTEGRITY MAINTAINED

REV 2

INSERT 3.3-64A

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3. For Functions 3 and 4, when a CHANNEL CALIBRATION channel is placed in an inoperable status solely for performance of ~~required testing~~ or maintenance, entry into associated Conditions and Required Actions may be delayed for up to ~~6 hours for a CHANNEL FUNCTIONAL TEST and for up to~~ 24 hours ~~for a CHANNEL CALIBRATION or maintenance,~~ provided the downscale trip of the inoperable channel is placed in the tripped condition.

REV. 2

INSERT 3.3-73A

P63

3. For Functions 3 and 4, when a channel is placed in an inoperable status solely for performance of ~~required testing~~ a CHANNEL CALIBRATION or maintenance, entry into associated Conditions and Required Actions may be delayed for up to ~~6 hours for a CHANNEL FUNCTIONAL TEST and for up to~~ 24 hours, for a ~~CHANNEL CALIBRATION or maintenance,~~ provided the downscale trip of the inoperable channel is placed in the tripped condition.

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INSERT 3.3-75A

INSERT 3.3-75B - Item A

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One of ~~three phase-to-phase~~ degraded voltage relay~~s~~ channel inoperable on a shutdown board.	A.1 ~~Add Insert A (R9~~ A.~~1~~2 Place the degraded voltage relay channel in trip.	15 days
B. One or more of the loss of voltage relay channels inoperable.	B.1 INSERT 3.3-75B-Item B B.~~1~~2 Place the inoperable loss of voltage relay channel(s) in trip.	10 days
C. Two ~~One~~ or more of the degraded voltage relay channels, or one or more associated timers, inoperable.	C.1 INSERT 3.3-75B-Item C C.~~1~~2 Place the inoperable degraded voltage relay channel(s) in trip.	10 days
D. Two or more ~~The~~ degraded voltage relay channels inoperable on one shutdown board. or one or more associated timers AND The loss of voltage relay channel(s) inoperable on the same shutdown board.	D.1 Verify by administrative means that the other shutdown boards and undervoltage relay channels, and associated timers are OPERABLE. AND D.2 Place the inoperable channels in trip.	Immediately 5 days

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INSERT 3.3 - 75B

Item A		
A.1	Verify by administrative means that the other two ~~phase-to-phase~~ degraded voltage relay~~s~~ channels and the loss of voltage relay channel on that shutdown board are OPERABLE.	Immediately
	And	
Item B		
B.1	Verify by administrative means that ~~the~~ two or more degraded voltage relay channels on that shutdown board ~~is~~ are OPERABLE, and associated timers,	Immediately
	And	
Item C		
C.1	Verify by administrative means that the loss of voltage relay channel~~s~~ on that shutdown board ~~is~~ are OPERABLE.	Immediately
	And	

Table 3.3.8.1-1 (page 1 of 1)
Loss of Power Instrumentation

FUNCTION	REQUIRED CHANNELS PER ~~BUS~~ Board	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage (Loss of Voltage)	(B2) This column		(B2) This column
a. ~~Bus~~ Board Undervoltage	~~{2}~~ 2	~~{SR 3.3.8.1.1}~~ ~~SR 3.3.8.1.2~~ SR 3.3.8.1.~~3~~2 SR 3.3.8.1.~~4~~3	≥ ~~{2800}~~ 2813 V and ≤ ~~{-}~~ 2927 V (Reset at)
Diesel Start Initiation b. Time Delay	~~{2}~~ 2	~~{SR 3.3.8.1.2}~~ SR 3.3.8.1.~~3~~2 SR 3.3.8.1.~~4~~3	≥ ~~{-}~~ 1.4 seconds and ≤ ~~{6.5}~~ 1.6 seconds
2. 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage (Degraded Voltage)			
a. ~~Bus~~ Board Undervoltage	~~{2}~~ 3	~~{SR 3.3.8.1.1}~~ ~~SR 3.3.8.1.2~~ SR 3.3.8.1.~~3~~1 SR 3.3.8.1.~~4~~3	≥ ~~{3280}~~ 3900 V and ≤ ~~{-}~~ 3940 V
b. Time Delay	~~{2}~~ 1	~~{SR 3.3.8.1.2}~~ SR 3.3.8.1.~~3~~2 SR 3.3.8.1.~~4~~3	≥ ~~{-}~~ 0.2 seconds and ≤ ~~{21.5}~~ 0.4 seconds

INSERT B3.3-77A

(P11) (except as marked)

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS BASES MARKUP

Replaced page 123 of 939 Rev. 1 (STS page B 3.3-7) with page 123 of 939 Rev. 2
Replaced page 123a of 939 Rev. 1 (insert page B 3.3-7A) with page 123a of 939 Rev. 2
Replaced page 141 of 939 Rev. 1 (STS page B 3.3-23) with page 141 of 939 Rev. 2
Added page 141a of 939 Rev. 2 (insert page B 3.3.1.1-C.1)
Replaced page 147 of 939 (STS page B 3.3-29) with page 147 of 939 Rev. 2
Replaced page 149 of 939 (STS page B 3.3-30) with page 149 of 939 Rev. 2
Replaced page 168 of 939 (STS page B 3.3-46) with page 168 of 939 Rev. 2
Replaced page 210d of 939 Rev. 1 (insert page B 3.3-79B) with page 210d of 939 Rev. 2
Replaced page 222 of 939 Rev. 1 (STS page B 3.3-91) with page 222 of 939 Rev. 2
Replaced page 224 of 939 Rev. 1 (STS page B 3.3-93) with page 224 of 939 Rev. 2
Replaced page 225 of 939 Rev. 1 (STS page B 3.3-94) with page 225 of 939 Rev. 2
Replaced page 233 of 939 Rev. 1 (STS page B 3.3-101) with page 233 of 939 Rev. 2
Replaced page 234 of 939 (insert page B 3.3-101A, 101B) with page 234 of 939 Rev. 2
Replaced page 238 of 939 (STS page B 3.3-104) with page 238 of 939 Rev. 2
Replaced page 239 of 939 Rev. 1 (STS page B 3.3-105) with page 239 of 939 Rev. 2
Replaced page 240 of 939 Rev. 1 (STS page B 3.3-106) with page 240 of 939 Rev. 2
Replaced page 244 of 939 Rev. 1 (STS page B 3.3-109) with page 244 of 939 Rev. 2
Replaced page 255 of 939 Rev. 1 (STS page B 3.3-118) with page 255 of 939 Rev. 2
Replaced page 261 of 939 Rev. 1 (STS page B 3.3-123) with page 261 of 939 Rev. 2
Replaced page 264 of 939 Rev. 1 (STS page B 3.3-125) with page 264 of 939 Rev. 2
Replaced page 266 of 939 Rev. 1 (STS page B 3.3-126) with page 266 of 939 Rev. 2
Replaced page 267 of 939 Rev. 1 (STS page B 3.3-127) with page 267 of 939 Rev. 2
Replaced page 271 of 939 Rev. 1 (STS page B 3.3-131) with page 271 of 939 Rev. 2
Replaced page 277 of 939 Rev. 1 (STS page B 3.3-137) with page 277 of 939 Rev. 2
Replaced page 324 of 939 Rev. 1 (STS page B 3.3-177) with page 324 of 939 Rev. 2
Replaced page 342 of 939 Rev. 1 (STS page B 3.3-192) with page 342 of 939 Rev. 2
Replaced page 372 of 939 (insert page B 3.3-216A) with page 372 of 939 Rev. 2
Replaced page 377 of 939 (STS page B 3.3-220) with page 377 of 939 Rev. 2
Replaced page 380 of 939 Rev. 1 (STS page B 3.3-222) with page 380 of 939 Rev. 2
Replaced page 382 of 939 (STS page B 3.3-223) with page 382 of 939 Rev. 2
Replaced page 383 of 939 Rev. 1 with page 383 of 939 Rev. 2
Replaced page 383a of 939 Rev. 1 with page 383a of 939 Rev. 2
Replaced page 384 of 939 Rev. 1 with page 384 of 939 Rev. 2
Replaced page 384a of 939 Rev. 1 with page 384a of 939 Rev. 2
Replaced page 390 of 939 (STS page B 3.3-228) with page 390 of 939 Rev. 2
Replaced page 395 of 939 Rev. 1 (STS page B 3.3-233) with page 395 of 939 Rev. 2

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2.a. Average Power Range Monitor Neutron Flux—High, Setdown (continued)

low power (i.e., MODE 2), the Average Power Range Monitor Neutron Flux—High, Setdown Function is capable of generating a trip signal that prevents fuel damage resulting from abnormal operating transients in this power range. For most operation at low power levels, the Average Power Range Monitor Neutron Flux—High, Setdown Function will provide a secondary scram to the Intermediate Range Monitor Neutron Flux—High Function because of the relative setpoints. With the IRMs at Range 9 or 10, it is possible that the Average Power Range Monitor Neutron Flux—High, Setdown Function will provide the primary trip signal for a corewide increase in power.

No specific safety analyses take direct credit for the Average Power Range Monitor Neutron Flux—High, Setdown Function. However, this Function indirectly ensures that before the reactor mode switch is placed in the run position, reactor power does not exceed 25% RTP (SL 2.1.1.1) when operating at low reactor pressure and low core flow. Therefore, it indirectly prevents fuel damage during significant reactivity increases with THERMAL POWER < 25% RTP.

The APRM System is divided into two groups of channels with three APRM channel inputs to each trip system. The system is designed to allow one channel in each trip system to be bypassed. Any one APRM channel in a trip system can cause the associated trip system to trip. Four channels of Average Power Range Monitor Neutron Flux—High, Setdown with two channels in each trip system are required to be OPERABLE to ensure that no single failure will preclude a scram from this Function on a valid signal. In addition, to provide adequate coverage of the entire core, at least ~~11~~ 14 LPRM inputs are required for each APRM channel, with at least two LPRM inputs from each of the four axial levels at which the LPRMs are located.

The Allowable Value is based on preventing significant increases in power when THERMAL POWER is < 25% RTP.

INSERT B33-7A1

The Average Power Range Monitor Neutron Flux—High, Setdown Function must be OPERABLE during MODE 2 when control rods may be withdrawn since the potential for criticality exists.

(continued)

INSERT B3.3-7A

The Average Power Range Monitor Neutron Flux--High, Setdown Function must be OPERABLE during MODE 5 with any control rod withdrawn from a core cell containing one or more fuel assemblies. The requirement for operability is not applicable if the Source Range Monitors (SRMs) are connected to give a noncoincidence, High Flux Scram, at 5×10^5 CPS. The SRMs can be configured to provide a High Flux Scram by removing eight shorting links. The ability of the SRMs to provide a High Flux Scram is in addition to the capability of the Intermediate Range Monitors (IRMs) and, thus, an exception to the requirement for the Average Power Range Monitor Neutron Flux--High, Setdown Function can be taken.

BASES

ACTIONS

B.1 and B.2 (continued)

system in trip would result in a scram ~~(or RPT)~~, B1
Condition D must be entered and its Required Action taken.

C.1

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same trip system for the same Function result in the Function not maintaining RPS trip capability. A Function is considered to be maintaining RPS trip capability when sufficient channels are OPERABLE or in trip (or the associated trip system is in trip), such that both trip systems will generate a trip signal from the given Function on a valid signal. ~~For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve—Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).~~ P15

~~For Function 8 (Turbine Stop Valve—Closure), this would require both trip systems to have three channels, each OPERABLE or in trip (or the associated trip system in trip).~~

Insert B3.3.1.1-C.1

~~Sufficient channels associated with the MSIVs in three main steam lines~~

~~Sufficient channels for Function 5 would require both trip systems to have at least one logic channel receiving inputs from two OPERABLE or tripped channels associated with different main steam lines.~~

~~Sufficient channels associated with three Turbine Stop Valves~~

~~Sufficient channels for Function 8 would require both trip systems to have at least one logic channel receiving inputs from two OPERABLE or tripped channels.~~

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

D.1

~~For Function 10 (Reactor Mode Switch-Shutdown Position) and Function 11 (Manual Scram), this would require the channel in both trip systems OPERABLE or in trip (or the associated trip system in trip).~~

Required Action D.1 directs entry into the appropriate Condition referenced in Table 3.3.1.1-1. The applicable Condition specified in the Table is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action

(continued)

INSERT B 3.3.1.1-C.1 ~~(THIS RETURNS BASES TO NUREG TEXT, NO JUSTIFICATION REQUIRED)~~

For the typical Function with one-out-of-two taken twice logic and the IRM and APRM Functions, this would require both trip systems to have one channel OPERABLE or in trip (or the associated trip system in trip). For Function 5 (Main Steam Isolation Valve - Closure), this would require both trip systems to have each channel associated with the MSIVs in three main steam lines (not necessarily the same main steam lines for both trip systems) OPERABLE or in trip (or the associated trip system in trip).

For Function 8 (Turbine Stop Valve - Closure), this would require both trip system to have three channels, each OPERABLE or in trip (or the associated trip system in trip).

BASES <(P1) except as noted>

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1.~~8~~5 and SR 3.3.1.1.~~9~~6 (continued)

A Frequency of 7 days is reasonable based on engineering judgment and the reliability of the IRMs and APRMs.

SR 3.3.1.1.~~8~~7 MWD/T adopted

LPRM gain settings are determined from the local flux profiles measured by the Traversing Incore Probe (TIP) System. This establishes the relative local flux profile for appropriate representative input to the APRM System. The 1000 ~~MWD/T~~ Frequency is based on operating experience with LPRM sensitivity changes.

~~effective full power hours~~ (P6)

<Unit 2+3 SR Nos>

<Unit 1 SR Nos.>

SR 3.3.1.1.~~9~~12 and SR 3.3.1.1.~~12~~16

SR 3.3.1.1.8 and SR 3.3.1.1.12

SR 3.3.1.1.8,

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function. Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology. The 92 day Frequency of SR 3.3.1.1.~~9~~8 is based on the reliability analysis of Reference 9.

(P60) INSERT B3.3-29A <Unit 2+3 only>

(P11) of SR 3.3.1.1.12

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

(P2) =>

~~SR 3.3.1.1.10~~

~~Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.1.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analysis. Under these conditions, the setpoint must be~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.1.1.10 (continued)~~

~~readjusted to be equal to or more conservative than accounted for in the appropriate setpoint methodology.~~

~~The Frequency of 92 days is based on the reliability analysis of Reference 9.~~

P2

SR 3.3.1.1.9,

P1

SR 3.3.1.1.1~~1~~0 and SR 3.3.1.1.13

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This 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 drifts between successive calibrations consistent with the plant specific setpoint methodology.

P11

INSERT B3.3-30A

Note 1 to SR 3.3.1.1.9 states that neutron detectors are excluded from 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 performing the 7 day calorimetric calibration (SR 3.3.1.1.2) and the 1000 ~~MWD/T~~ MWD/T LPRM calibration against the TIPs (SR 3.3.1.1.~~8~~9). A second Note for SR 3.3.1.1.9 is provided that requires the APRM and IRM SRs to be performed within 12 hours of entering MODE 2 from MODE 1. Testing of the MODE 2 APRM and IRM Functions cannot be performed in MODE 1 without utilizing jumpers, lifted leads, or movable links. This Note allows entry into MODE 2 from MODE 1 if the associated Frequency is not met per SR 3.0.2. Twelve hours is based on operating experience and in consideration of providing a reasonable time in which to complete the SR.

~~effective full power hours~~

P6

7

P1

P91

The frequency of SR 3.3.1.1.9 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

The Frequency of SR 3.3.1.1.1~~1~~0 is based upon the assumption of a 184 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. The Frequency of SR 3.3.1.1.13 is based upon the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

P1

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1. Rod Block Monitor (continued)

The RBM Function satisfies Criterion 3 of the NRC Policy Statement. (Rq. 10) P8

Two channels of the RBM are required to be OPERABLE, with their setpoints within the appropriate Allowable Value ~~for the associated power range~~ P90 to ensure that no single instrument failure can preclude a rod block from this Function. The ~~actual~~ setpoints are calibrated consistent with applicable setpoint methodology. P11 (nominal trip setpoint)

Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected to ensure that the setpoints do not exceed the Allowable Values between successive CHANNEL CALIBRATIONS. Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable. Trip setpoints are those predetermined values of output at which an action should take place. The setpoints are compared to the actual process parameter (e.g., reactor power), and when the measured output value of the process parameter exceeds the setpoint, the associated device (e.g., trip unit) changes state. The analytic limits are derived from the limiting values of the process parameters obtained from the safety analysis. The Allowable Values are derived from the analytic limits, corrected for calibration, process, and some of the instrument errors. The trip setpoints are then determined accounting for the remaining instrument errors (e.g., drift). The trip setpoints derived in this manner provide adequate protection because instrumentation uncertainties, process effects, calibration tolerances, instrument drift, and severe environment ~~errors~~ effects (for channels that must function in harsh environments as defined by 10 CFR 50.49) are accounted for. P11 a1

The RBM is assumed to mitigate the consequences of an RWE event when operating ≥ 29% RTP. Below this power level, the consequences of an RWE event will not exceed the MCPR SL and, therefore, the RBM is not required to be OPERABLE (Ref. 3). When operating < 90% RTP, analyses (Ref. 3) have shown that with an initial MCPR ≥ ~~1.70~~ 1.75, no RWE event will result in exceeding the MCPR SL. Also, the analyses demonstrate that when operating at ≥ 90% RTP with MCPR ≥ ~~1.40~~ 1.44, no RWE event will result in exceeding the MCPR

(continued)

Table B 3.3.3.2-1 (Page 1 of 3)
Backup Control System Instrumentation and Controls

	FUNCTION	NUMBER REQUIRED
Instrument Parameter		
1	Reactor Water Level Indication	1
2	Reactor Pressure Indication	1
3	Suppression Pool Temperature Indication	1
4	Suppression Pool Level Indication	1
5	Drywell Pressure Indication	1
6	RHR Flow Indication	1
7	RCIC Flow Indication	1, note a
8	RCIC Turbine Speed Indication	1
9	Drywell Temperature Indication	units 1 and 2 — 1 note o — unit 3
10	RHRSW Header Pressure	1, note p
Transfer/Control Parameter		
11	Main Steam Relief Valve (MSRV) Transfer & Control	3, note b
12	Main Steam Isolation Valve (MSIV) Transfer & Control (Closure)	4, note c
13	Main Steam Drain Line Isolation Valve	1, note d
14	RHRSW Pumps	note e
15	RHRSW Discharge Valves for RHR Loop I Heat Exchangers	2, note f

note a: RCIC flow indication may be obtained from the Flow Indicating Controller
note b: 1 required for each of 3 MSRVs.
note c: 1 MSIV required per penetration, may be either inboard valve or outboard valve.
note d: 1 Main Steam Drain Line isolation valve required, may be either inboard valve or outboard valve.
note e: There are 12 RHRSW pumps. All are equipped with emergency transfer switches. 2 of the 12 must be available for EECW service (supports all units)and an additional 1 must be available for RHRSW service.
note f: 1 Discharge Valve per RHR Loop I Heat Exchanger for a total of 2
note o: Drywell Temperature Indication not required for Unit 3 ← note not used (units 1 and 2 only)
note p: The RHRSW Pressure indicator for the Header of the RHRSW Pump that supports RHR service is required.

unit 3

B 3.3 INSTRUMENTATION

B 3.3.4.2 Anticipated Transient Without Scram Recirculation Pump Trip (ATWS-RPT) Instrumentation

BASES

BACKGROUND

The ATWS-RPT System initiates an RPT, adding negative reactivity, following events in which a scram does not (but should) occur, to lessen the effects of an ATWS event. Tripping the recirculation pumps adds negative reactivity from the increase in steam voiding in the core area as core flow decreases. When Reactor Vessel Water Level — ~~Low Low, Level 2~~ or Reactor Steam Dome Pressure — High setpoint is reached, the recirculation pump ~~drive~~ motor breakers trip. (P11) (Low Low, Level 2)

The ATWS-RPT System (Ref. 1) includes sensors, relays, bypass capability, circuit breakers, and switches that are necessary to cause initiation of an RPT. The channels include electronic equipment (e.g., trip units) that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an ATWS-RPT signal to the trip logic. (P5) (Low Low, Level 2)

The ATWS-RPT consists of two independent trip systems, with two channels of Reactor Steam Dome Pressure — High and two channels of Reactor Vessel Water Level — ~~Low Low, Level 2~~ in each trip system. Each ATWS-RPT trip system is a two-out-of-two logic for each Function. Thus, either two Reactor Water Level — ~~Low Low, Level 2~~ or two Reactor Pressure — High signals are needed to trip a trip system. The outputs of the channels in a trip system are combined in a logic so that either trip system will trip both recirculation pumps (by tripping the respective ~~drive~~ motor breakers). (P11) (VESSEL) (P11)

There ~~is one drive~~ are two motor breakers provided for each of the two recirculation pumps for a total of ~~two~~ four breakers. The output of each trip system is provided to ~~both recirculation pump breakers~~ ONE OF THE TWO BREAKERS FOR EACH RECIRCULATION PUMP. (P11) (P11)

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Reactor Protection System by providing a diverse trip to mitigate the consequences of a postulated ATWS event. The Reactor Steam Dome Pressure—High and Reactor Vessel Water Level—~~Low Low, Level 2~~ Low Low, Level 2 Functions are required to be OPERABLE in MODE 1, since the reactor is producing significant power and the recirculation system could be at high flow. During this MODE, the potential exists for pressure increases or low water level, assuming an ATWS event. In MODE 2, the reactor is at low power and the recirculation system is at low flow; thus, the potential is low for a pressure increase or low water level, assuming an ATWS event. Therefore, the ATWS-RPT is not necessary. In MODES 3 and 4, the reactor is shut down with all control rods inserted; thus, an ATWS event is not significant and the possibility of a significant pressure increase or low water level is negligible. In MODE 5, the one rod out interlock ensures that the reactor remains subcritical; thus, an ATWS event is not significant. In addition, the reactor pressure vessel (RPV) head is not fully tensioned and no pressure transient threat to the reactor coolant pressure boundary (RCPB) exists.

The specific Applicable Safety Analyses and LCO discussions are listed below on a Function by Function basis.

a. Reactor Vessel Water Level—~~Low Low, Level 2~~ Low Low, Level 2 (LS-3-58A1, LS-3-58B1, LS-3-58C1, AND LS-3-58D1)

Low RPV water level indicates the capability to cool the fuel may be threatened. Should RPV water level decrease too far, fuel damage could result. Therefore, the ATWS-RPT System is initiated at Level 2 to aid in maintaining level above the top of the active fuel. The reduction of core flow reduces the neutron flux and THERMAL POWER and, therefore, the rate of coolant boiloff.

Reactor vessel water level signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

Four channels of Reactor Vessel Water Level—~~Low Low, Level 2~~ Low Low, Level 2, with two channels in each trip system, are available and required to be OPERABLE to ensure that

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Reactor Vessel Water Level ~~Low Low, Level 2~~ (P11)
(continued)

no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Vessel Water Level ~~Low Low, Level 2~~ (Low Low, Level 2) Allowable Value (P11) is chosen so that the system will not be initiated after a Level 3 scram with feedwater still available, and for convenience with the reactor core isolation cooling initiation.

b. Reactor Steam Dome Pressure—High (PIS-3-204A, PIS-3-204B, PIS-3-204C, AND PIS-3-204D) (P11)

Excessively high RPV pressure may rupture the RCPB. An increase in the RPV pressure during reactor operation compresses the steam voids and results in a positive reactivity insertion. This increases neutron flux and THERMAL POWER, which could potentially result in fuel failure and overpressurization. The Reactor Steam Dome Pressure—High Function initiates an RPT for transients that result in a pressure increase, counteracting the pressure increase by rapidly reducing core power generation. For the overpressurization event, the RPT aids in the termination of the ATWS event and, along with the safety/relief valves, limits the peak RPV pressure to less than the ASME Section III Code ~~Service Level C~~ limits ~~(1500 psig)~~. (P7)

The Reactor Steam Dome Pressure—High signals are initiated from four pressure transmitters that monitor reactor steam dome pressure. Four channels of Reactor Steam Dome Pressure—High, with two channels in each trip system, are available and are required to be OPERABLE to ensure that no single instrument failure can preclude an ATWS-RPT from this Function on a valid signal. The Reactor Steam Dome Pressure—High Allowable Value is chosen to provide an adequate margin to the ASME Section III Code ~~Service Level C~~ limits (P7) ~~allowable Reactor Coolant System pressure.~~

ACTIONS

A Note has been provided to modify the ACTIONS related to ATWS-RPT instrumentation channels. Section 1.3, Completion

(continued)

B 3.3 INSTRUMENTATION

B 3.3.5.1 Emergency Core Cooling System (ECCS) Instrumentation

BASES

BACKGROUND

The purpose of the ECCS instrumentation is to initiate appropriate responses from the systems to ensure that the fuel is adequately cooled in the event of a design basis accident or transient.

Portions of the ECCS instrumentation also provide for the generation of the Common Accident Signal which initiates the DG's and EECW system. Refer to LCO 3.8.1, "AC Sources - Operating", for Operability requirements of the Common Accident Signal Logic.

For most anticipated operational occurrences and Design Basis Accidents (DBAs), a wide range of dependent and independent parameters are monitored.

The ECCS instrumentation actuates core spray (CS), low pressure coolant injection (LPCI), high pressure coolant injection (HPCI), Automatic Depressurization System (ADS), and the ~~diesel generators (DGs)~~. The equipment involved with each of these systems is described in the Bases for LCO 3.5.1, "ECCS—Operating."

Core Spray System

Each pump can be controlled manually by a control room remote switch

Reactor Water level and drywell pressure are each

The CS System may be initiated by ~~either~~ automatic ~~or manual~~ means. Automatic initiation occurs for conditions of Reactor Vessel Water Level—Low Low Low, Level 1 and/or Drywell Pressure—High and Reactor Steam Dome Pressure-Low. ~~Each of these diverse variables is~~ both monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of these ~~eight~~ trip units are connected to relays whose contacts are arranged in a one-out-of-two taken twice logic (i.e., two trip systems) for each Function.

INSERT B33-101A

The high drywell pressure and low reactor water level initiation signals are sealed in signals and must be manually reset. The CS System can be reset if reactor water level has been restored, even if the high drywell pressure condition persists. The logic can also be initiated by use of a manual push button (one push button per subsystem). Upon receipt of an initiation signal, ~~the CS pumps are started immediately after power is available~~.

INSERT B 3.3-101B

The CS test line isolation valve ~~, which is also a primary containment isolation valve (PCIV),~~ is closed on a CS initiation signal to allow full system flow assumed in the

(continued)

INSERT B3.3-101A

P11

The Reactor Steam Dome Pressure-Low variable is monitored by ~~four~~ ~~two~~ two transmitters for each ~~subsystem~~ trip system. The outputs from these transmitters are connected to relays arranged in a one-out-of-two logic.

INSERT B3.3-101B

if normal AC power is available, the four core spray pumps start one at a time, in order, at 0, 7, 14, and 21 seconds. If normal AC power is not available, the four core spray pumps start seven seconds after standby power becomes available. (The LPCI pumps start as soon as standby power is available.)

BASES

BACKGROUND High Pressure Coolant Injection System (continued)

connected to relays whose contacts are arranged in a one-out-of-two taken twice logic for each Function.

P11

The HPCI pump discharge flow is monitored by a flow switch ~~transmitter~~. Upon automatic initiation, when the pump is running and discharge flow is low enough so that pump overheating may occur, the minimum flow return line valve is opened. The valve is automatically closed if flow is above the minimum flow setpoint to allow the full system flow, ~~assumed in the accident analysis~~. HOWEVER, FLOW RATES ASSUMED IN THE ACCIDENT ANALYSIS CAN BE ACHIEVED WITH THE MINIMUM FLOW VALVE OPEN. P11 The HPCI test line isolation valve ~~(which is also a PCIV)~~ is closed upon receipt of a HPCI initiation signal to allow the full system flow assumed in the accident analysis and maintain primary containment isolated ~~in the~~ event HPCI is not operating. P11

P11

The HPCI System also monitors the water levels in the condensate storage tank (CST) and the suppression pool because these are the two sources of water for HPCI operation. Reactor grade water in the CST is the normal source. Upon receipt of a HPCI initiation signal, the CST suction valve is automatically signaled to open (it is normally in the open position) unless both suppression pool suction valves are open. If the water level in the HPCI pump supply header from the CST falls below a preselected level, first the suppression pool suction valves automatically open, and then the CST suction valve automatically closes. Two level switches are used to detect low water level in the CST. Either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The suppression pool suction valves also automatically open and the CST suction valve closes if high water level is detected in the suppression pool. To prevent losing suction to the pump, the suction valves are interlocked so that one suction path must be open before the other automatically closes.

The HPCI provides makeup water to the reactor until the reactor vessel water level reaches the Reactor Vessel Water Level—High, Level 8 trip, at which time the HPCI turbine trips, which causes the turbine's stop valve ~~and the injection valves~~ to close. The logic is two-out-of-two to provide high reliability of the HPCI System. The HPCI P11

(continued)

BASES

BACKGROUND

High Pressure Coolant Injection System (continued)

System automatically restarts if a Reactor Vessel Water Level — Low Low, Level 2 signal is subsequently received.

Automatic Depressurization System

The ADS may be initiated by either automatic or manual means. Automatic initiation occurs when signals indicating Reactor Vessel Water Level — Low Low Low, Level 1; Drywell Pressure — High or ADS Bypass ~~Low Water Level Actuation~~ Timer; confirmed Reactor Vessel Water Level — Low, Level 3; and CS or LPCI Pump Discharge Pressure — High are all present and the ADS Initiation Timer has timed out. There are two transmitters each for Reactor Vessel Water Level — Low Low Low, Level 1 and Drywell Pressure — High, and one transmitter for confirmed Reactor Vessel Water Level — Low, Level 3 in each of the two ADS trip systems. Each of these transmitters connects to a trip unit, which then drives a relay whose contacts form the initiation logic.

P11 High Drywell Pressure

(confirmatory) P11

P11 ADS Initiation

Each ADS trip system includes a time delay between satisfying the initiation logic and the actuation of the ADS valves. The ADS Initiation Timer time delay setpoint chosen is long enough that the HPCI has sufficient operating time to recover to a level above Level 1, yet not so long that the LPCI and CS Systems are unable to adequately cool the fuel if the HPCI fails to maintain that level. An alarm in the control room is annunciated when either of the timers is timing. Resetting the ADS initiation signals resets the ADS Initiation Timers.

All four LPCI Pumps and one discharge pressure permissive switch from all four CS Pumps.

P11

CS pumps (A or B and either C or D) or

and one discharge pressure permissive switch for each LPCI pump

four

~~two of the four CS pumps (A and B for one trip system and C and D for the other trip system)~~

The ADS also monitors the discharge pressures of the four LPCI pumps and the ~~two~~ CS pumps. Each ADS trip system includes two discharge pressure permissive ~~transmitters~~ switches from ~~both CS and from two LPCI pumps in the associated Division (i.e., Division 1 LPCI subsystems A and B input to ADS trip system A, and Division 2 LPCI subsystems B and C input to ADS trip system B)~~. The signals are used as a permissive for ADS actuation, indicating that there is a source of core coolant available once the ADS has depressurized the vessel. Any one of the ~~six low pressure~~ four LPCI pumps is sufficient to permit automatic depressurization.

BASES

BACKGROUND

and Pump Discharge Pressure—High. P11

Automatic Depressurization System (continued)

to initiate an ADS trip system.

The ADS logic in each trip system is arranged in two strings. Each string has a contact from each of the following variables: Reactor Vessel Water Level — Low Low Low, Level 1; Drywell Pressure — High; ~~or Low Water Level Actuation~~ Timer. One of the two strings in each trip system must also have a confirmed Reactor Vessel Water Level — Low, Level 3. ~~All~~ contacts in both logic strings must close, and the ADS initiation timer must time out. ~~and a CS or LPCI pump discharge pressure signal must be present to initiate an ADS trip system.~~ Either the A or B trip system will cause all the ADS relief valves to open. Once the Drywell Pressure — High signal, the ADS ~~Low Water Level Actuation~~ High Drywell Pressure Bypass Timer, or the ADS initiation signal is present, it is individually sealed in until manually reset. P11

High Drywell Pressure Bypass

P11 (Confirmatory)

Either the Drywell Pressure-High or the High Drywell Pressure Bypass Timer contacts and all remaining

Manual inhibit switches are provided in the control room for the ADS; however, their function is not required for ADS OPERABILITY (provided ADS is not inhibited when required to be OPERABLE).

Diesel Generators

P11

The DGs may be initiated by either automatic or manual means. Automatic initiation occurs for conditions of Reactor Vessel Water Level — Low Low Low, Level 1 or Drywell Pressure — High and Reactor Steam Dome Pressure-Low. The DGs are also initiated upon loss of voltage signals. (Refer to the Bases for LCO 3.3.8.1, "Loss of Power (LOP) Instrumentation," for a discussion of these signals.) Each of these diverse variables is monitored by four redundant transmitters, which are, in turn, connected to four trip units. The outputs of the four trip units are connected to relays whose contacts are connected to a one-out-of-two taken twice logic to initiate all ~~three~~ eight DGs ~~(2A, 1B, and 2C)~~. The DGs receive their initiation signals from the CS System initiation logic. The DGs can also be started manually from the control room and locally from the associated DG room. The DG initiation signal is a sealed in signal and must be manually reset. The DG initiation logic is reset by resetting the associated ECCS initiation logic. Upon receipt of a loss of coolant accident (LOCA) initiation signal, each DG is automatically started, is ready to load in approximately ~~12~~ 10 seconds, and will run in standby conditions (rated voltage and speed, with the DG output

P99 ~~P99~~ P11 A, B, C, D, 3A, 3B, 3C and 3D P11 P11 both

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

1.a, 2.a. Reactor Vessel Water Level — Low Low Low, Level 1 (continued)

Reactor Vessel Water Level — Low Low Low, Level 1 signals are initiated from four level transmitters that sense the difference between the pressure due to a constant column of water (reference leg) and the pressure due to the actual water level (variable leg) in the vessel.

The Reactor Vessel Water Level — Low Low Low, Level 1 Allowable Value is chosen to allow time for the low pressure ~~core flooding systems~~ injection/spray subsystems to activate and provide adequate cooling.

Four channels of Reactor Vessel Water Level — Low Low Low, Level 1 Function are only required to be OPERABLE when the ECCS ~~, or DG(s) are~~ is required to be OPERABLE to ensure that no single instrument failure can preclude ECCS ~~, and DG~~ or EECW System initiation. Refer to LCO 3.5.1 and LCO 3.5.2, "ECCS — Shutdown," for Applicability Bases for the low pressure ECCS subsystems; ~~LCO 3.8.1, "AC Sources — Operating"; and LCO 3.8.2, "AC Sources — Shutdown," for Applicability Bases for the DGs.~~ INSERT B 3.3-109A

1.b, 2.b. Drywell Pressure — High (PIS-64-58A-D)

High pressure in the drywell could indicate a break in the reactor coolant pressure boundary (RCPB). The low pressure ECCS ~~, and associated DGs are~~ is initiated upon receipt of the Drywell Pressure — High Function in order to minimize the possibility of fuel damage. The Drywell Pressure — High Function, along with the Reactor ~~Water Level — Low Low Low, Level 1~~ Steam Pressure-Low Function, ~~is~~ are directly assumed in the analysis of the recirculation line break (Ref. 2A). The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

The Drywell Pressure-High is also utilized in the deployment of the Common Accident signal which initiates the DGs and EECW System. (Refer to LCO 3.8.1, "AC Sources - Operating" for Operability requirements of the Common Accident Signal Logic).

High drywell pressure signals are initiated from four pressure transmitters that sense drywell pressure. The Allowable Value was selected to be as low as possible and be indicative of a LOCA inside primary containment.

The Drywell Pressure — High Function is required to be OPERABLE when the ECCS ~~, or DG, is~~ or EECW System are required to be OPERABLE in conjunction with times when the primary containment is

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

3.e. Suppression Pool Water Level—High (continued)

This Function is implicitly assumed in the accident and transient analyses (which take credit for HPCI) since the analyses assume that the HPCI suction source is the suppression pool.

Suppression Pool Water Level—High signals are initiated from two level switches. The logic is arranged such that either switch can cause the suppression pool suction valves to open and the CST suction valve to close. The Allowable Value for the Suppression Pool Water Level—High Function is chosen to ensure that HPCI will be aligned for suction from the suppression pool before the water level reaches the point at which suppression pool design loads would be exceeded.

One ~~Two~~ channel~~s~~ of Suppression Pool Water Level—High Function is~~are~~ required to be OPERABLE only when HPCI is required to be OPERABLE ~~to ensure that no single instrument failure can preclude HPCI swap to suppression pool source~~. Refer to LCO 3.5.1 for HPCI Applicability Bases.

3.f. High Pressure Coolant Injection Pump Discharge Flow—Low (Bypass) (FIS-73-33)

The minimum flow instrument~~s~~ are provided to protect the HPCI pump from overheating when the pump is operating at reduced flow ~~and the associated injection valve is not fully open~~. The minimum flow line valve is opened when low flow is sensed, and the valve is automatically closed when the flow rate is adequate to protect the pump. The High Pressure Coolant Injection Pump Discharge Flow—Low Function ~~is assumed to be OPERABLE and capable of closing~~ the minimum flow valve ~~to ensure that the ECCS flow assumed~~ during the transients and accidents analyzed in References ~~1,~~ 2 and 3 are met. The core cooling function of the ECCS, along with the scram action of the RPS, ensures that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

One flow ~~transmitter~~ switch is used to detect the HPCI System's flow rate. The logic is arranged such that the ~~transmitter~~ switch causes the minimum flow valve to open. The logic will close the minimum flow valve once the closure setpoint is exceeded.

(continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4.e, 4.f, 5.e, 5.f. Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure—High (continued)

one LPCI pump ~~(both channels for the pump)~~ or two CS pumps (CS Pumps A or B and either C or D) indicate the high discharge pressure condition. The Pump Discharge Pressure—High Allowable Value is less than the pump discharge pressure when the pump is operating in a full flow mode and high enough to avoid any condition that results in a discharge pressure permissive when the CS and LPCI pumps are aligned for injection and the pumps are not running. The actual operating point of this function is not assumed in any transient or accident analysis.

INSERT B 3.3-123A

Twelve channels of Core Spray and Low Pressure Coolant Injection Pump Discharge Pressure—High Function are only required to be OPERABLE when the ADS is required to be OPERABLE to ensure that no single instrument failure can preclude ADS initiation. ~~Two~~ Four CS channels associated with CS pumps A through D and ~~four~~ eight LPCI channels associated with LPCI pumps A ~~and~~ through D are required for trip system A. Two CS channels associated with CS pump B and four LPCI channels associated with LPCI pumps B and C are required for trip system B. Refer to LCO 3.5.1 for both ADS Applicability Bases.

4.g, 5.g. Automatic Depressurization System ~~Low Water Level Actuation~~ High Drywell Pressure Bypass Timer

One of the signals required for ADS initiation is Drywell Pressure—High. However, if the event requiring ADS initiation occurs outside the drywell (e.g., main steam line break outside containment), a high drywell pressure signal may never be present. Therefore, the Automatic Depressurization System ~~Low Water Level Actuation~~ Timer is used to bypass the Drywell Pressure—High Function after a certain time period has elapsed. Operation of the Automatic Depressurization System ~~Low Water Level Actuation~~ Timer Function is not assumed in any accident analysis. The instrumentation was installed to meet requirements of NUREG-0737, Item II.K.3.18 (Ref. 6) and is retained in the TS because ADS is part of the primary success path for mitigation of a DBA.

P9 There are ~~two~~ four Automatic Depressurization System ~~Low Water Level Actuation~~ Timer relays, ~~two~~ two in each of the two ADS trip systems. The Allowable Value for the Automatic Depressurization System ~~Low Water Level Actuation~~ Timer is chosen to ensure that there is still time after

(continued)

ACTIONS (continued)

not within limits will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable ECCS instrumentation channels provide appropriate compensatory measures for separate inoperable Condition entry for each inoperable ECCS instrumentation channel.

A.1

Required Action A.1 directs entry into the appropriate Condition referenced in Table 3.3.5.1-1. The applicable Condition referenced in the table is Function dependent. Each time a channel is discovered inoperable, Condition A is entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

Required Actions B.1 and B.2 are intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action B.1 features would be those that are initiated by Functions 1.a, 1.b, 2.a, and 2.b (e.g., low pressure ECCS). The Required Action B.2 system would be HPCI. For Required Action B.1, redundant automatic initiation capability is lost if (a) two or more Function 1.a channels are inoperable and untripped ~~in the same trip system~~, (b) two or more Function 2.a channels are inoperable and untripped ~~in the same trip system,~~ (c) two or more Function 1.b channels are inoperable and untripped ~~in the same system, or~~ or (d) two or more Function 2.b channels are inoperable and untripped ~~in the same trip system.~~ such that both trip systems lose initiation capability. For low pressure ECCS, since each inoperable channel would have Required Action B.1 applied separately (refer to ACTIONS Note), each inoperable channel would only require the affected portion of the associated system of low pressure ECCS ~~and DGs~~ to be declared inoperable. However, since channels in both associated low pressure ECCS subsystems (e.g., both CS subsystems) are inoperable and untripped, and the Completion Times started concurrently for the channels in both subsystems, this results in the affected portions in the associated low

INSERT B.3.3-125A

(continued)

BASES (P11) <except as marked>

ACTIONS

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

(P49) (P48) pressure ECCS ~~and DGs~~ being concurrently declared inoperable.

For Required Action B.2, redundant (or more) automatic HPCI initiation capability is lost if two Function 3.a or two Function 3.b channels are inoperable and untripped ~~in the same trip system~~. In this situation (loss of redundant automatic HPCI initiation capability), the 24 hour allowance of Required Action B.3 is not appropriate and the ~~feature(s) associated with the inoperable, untripped channels~~ such that the trip system loses initiation capability must be declared inoperable within 1 hour. As noted (Note 1 to Required Action B.1), Required Action B.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the low pressure ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 24 hours (as allowed by Required Action B.3) is allowed during MODES 4 and 5. There is no similar Note provided for Required Action B.2 since HPCI instrumentation is not required in MODES 4 and 5; thus, a Note is not necessary. HPCI System

Notes are also provided (Note 2 to Required Action B.1 and the Note to Required Action B.2) to delineate which Required Action is applicable for each Function that requires entry into Condition B if an associated channel is inoperable. This ensures that the proper loss of initiation capability check is performed. Required Action B.1 (the Required Action for certain inoperable channels in the low pressure ECCS subsystems) is not applicable to Function 2.e, since this Function provides backup to administrative controls ensuring that operators do not divert LPCI flow from injecting into the core when needed. Thus, a total loss of Function 2.e capability for 24 hours is allowed, since the LPCI subsystems remain capable of performing their intended function.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock." For Required Action B.1, the Completion Time only begins upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable, untripped channels within the same

(continued)

BASES (P11) <except as marked>

ACTIONS

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

Function as described in the paragraph above. For Required Action B.2, the Completion Time only begins upon discovery that the HPCI System cannot be automatically initiated due to ~~two~~ inoperable, untripped channels ~~for the associated Function in the same trip system.~~ The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

Within the same Function as described in the paragraph above.

Because of the diversity of sensors available to provide initiation signals and the redundancy of the ECCS design, an allowable out of service time of 24 hours has been shown to be acceptable (Ref. ~~8~~ 4) to permit restoration of any inoperable channel to OPERABLE status. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action B.3. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure, and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the inoperable channel in trip would result in an initiation), Condition H must be entered and its Required Action taken.

(i.e. at least one CS pump in both subsystems is affected)

(d) one or more Function 1.e channels are inoperable in both trip systems

~~affecting CS pumps in different subsystems~~

C.1 and C.2

Required Action C.1 is intended to ensure that appropriate actions are taken if multiple, inoperable channels within the same Function result in redundant automatic initiation capability being lost for the feature(s). Required Action C.1 features would be those that are initiated by Functions ~~1.e, 2.c~~ 1.c, 2.d, and 2.f (i.e., low pressure ECCS). Redundant automatic initiation capability is lost if either (a) ~~two~~ four or more Function 1.e channels are inoperable ~~in the same trip system~~, (b) ~~two Function 2.c channels are inoperable in the same trip system,~~ (c) ~~two~~ or more Function 2.d channels are inoperable ~~in the same trip system~~, or (d) two or more Function 2.f channels are inoperable. In this situation (loss of redundant automatic initiation capability), the 24 hour allowance of Required Action C.2 is not appropriate and the feature(s) associated with the inoperable channels must be declared inoperable within 1 hour. Since each

such that both trip systems loses initiation capability

Such that both trip systems lose initiation capability

Such that the trip systems cannot start both LPCI pumps in at least one subsystem.

~~affecting two LPCI pumps~~

(continued)

BASES

ACTIONS

E.1 and E.2 (continued)

all four CS P82

~~low pressure ECCS~~ pumps ~~being~~ concurrently declared inoperable. Stet

P15

P11

In this situation (loss of minimum flow ~~redundant automatic initiation~~ capability), the 7 day allowance of Required Action E.2 is not appropriate and the subsystem associated with each inoperable channel must be declared inoperable within 1 hour. As noted (Note 1 to Required Action E.1), Required Action E.1 is only applicable in MODES 1, 2, and 3. In MODES 4 and 5, the specific initiation time of the ECCS is not assumed and the probability of a LOCA is lower. Thus, a total loss of initiation capability for 7 days (as allowed by Required Action E.2) is allowed during MODES 4 and 5. A Note is also provided (Note 2 to Required Action E.1) to delineate that Required Action E.1 is only applicable to ~~low pressure ECCS Functions~~ Function 1.d. Required Action E.1 is not applicable to HPCI Function 3.f since the loss of one channel results in a loss of the Function (one-out-of-one logic). This loss was considered during the development of Reference ~~8~~ 4 and considered acceptable for the 7 days allowed by Required Action E.2.

Move to Previous Page

for both Subsystems

P11

P8

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. This Completion Time also allows for an exception to the normal "time zero" for beginning the allowed outage time "clock."

For Required Action E.1, the Completion Time only begins upon discovery that a redundant feature in the same system (e.g., both CS subsystems) cannot be automatically initiated due to inoperable channels within the same Function as described in the paragraph above. The 1 hour Completion Time from discovery of loss of initiation capability is acceptable because it minimizes risk while allowing time for restoration of channels.

CS P82 P82 CS

If the instrumentation that controls the pump minimum flow valve is inoperable, such that the valve will not automatically open, extended pump operation with no injection path available could lead to pump overheating and failure. If there were a failure of the instrumentation, such that the valve would not automatically close, a portion of the pump flow could be diverted from the reactor vessel injection path, causing insufficient core cooling. These

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1.3

Calibration of trip units provides a check of the actual trip setpoints. The channel must be declared inoperable if the trip setting is discovered to be less conservative than the Allowable Value specified in Table 3.3.5.1-1. If the trip setting is discovered to be less conservative than accounted for in the appropriate setpoint methodology, but is not beyond the Allowable Value, the channel performance is still within the requirements of the plant safety analyses. Under these conditions, the setpoint must be readjusted to be equal to or more conservative than the setting accounted for in the appropriate setpoint methodology.

P2

The Frequency of 92 days is based on the reliability analysis of Reference 5.

P1

SR 3.3.5.1.3,

SR 3.3.5.1.4 and SR 3.3.5.1.5

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. This test verifies the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drifts between successive calibrations consistent with the plant specific setpoint methodology.

The Frequency of SR 3.3.5.1.4 is based upon the assumption of a 92 day calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

P11

The Frequency~~y~~ies of SR 3.3.5.1.5 ~~is~~ are based upon ~~the assumption of an 18 month calibration interval in the determination of~~ the magnitude of equipment drift in the setpoint analysis.

SR 3.3.5.1.3, SR 3.3.5.1.4, and

SR 3.3.5.1.6

The LOGIC SYSTEM FUNCTIONAL TEST demonstrates the OPERABILITY of the required initiation logic for a specific channel. The system functional testing performed in LCO 3.5.1, LCO 3.5.2, LCO 3.8.1, ~~and LCO 3.8.2~~ overlaps this Surveillance to complete testing of the assumed safety function. P11

LCO 3.7.2, and

P11 PAGE 277 OF 939

(continued)

BASES

ACTIONS

B.1 (continued)

risk while allowing time for restoration or tripping of channels.

INSERT B 3.3-177A

P25

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.6.1-1. The applicable Condition specified in Table 3.3.6.1-1 is Function and MODE or other specified condition dependent and may change as the Required Action of a previous Condition is completed. Each time an inoperable channel has not met any Required Action of Condition A or B and the associated Completion Time has expired, Condition C will be entered for that channel and provides for transfer to the appropriate subsequent Condition.

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

~~affected penetration flow path(s)~~

P03

~~the affected penetration flow path(s)~~

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours (Required Actions D.2.1 and D.2.2). Alternately, the associated MSLs may be isolated (Required Action D.1), and, if allowed (i.e., plant safety analysis allows operation with an MSL isolated), operation with that MSL isolated may continue. Isolating the affected MSL accomplishes the safety function of the inoperable channel. The Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

STET

P25 ~~penetration flow path(s)~~

E.1

If the channel is not restored to OPERABLE status or placed in trip within the allowed Completion Time, the plant must be placed in a MODE or other specified condition in which the LCO does not apply. This is done by placing the plant in at least MODE 2 within 6 hours.

(continued)

BASES

ACTIONS (continued)

B.1

Required Action B.1 is intended to ensure that appropriate actions are taken if multiple, inoperable, untripped channels within the same Function result in a complete loss of automatic isolation capability for the associated secondary containment penetration flow path(s) or a complete loss of automatic initiation capability for the SGT System. A Function is considered to be maintaining secondary containment isolation capability when sufficient channels are OPERABLE or in trip, such that at least one trip system will generate a trip signal from the given Function on a valid signal. This ensures that one of the two SCIVs in the associated penetration flow path(s) and one SGT subsystem can be initiated on an isolation signal from the given Function. For the Functions with two one-out-of-two logic trip systems (Functions 1, 2, ~~3, and 4~~), this would require one trip system to have one channel OPERABLE or in trip. The Condition does not include the Manual Initiation Function (Function 5), since it is not assumed in any accident or transient analysis. Thus, a total loss of manual initiation capability for 24 hours (as allowed by Required Action A.1) is allowed.

The Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time is acceptable because it minimizes risk while allowing time for restoration or tripping of channels.

C.1.1, C.1.2, C.2.1, and C.2.2

If any Required Action and associated Completion Time of Condition A or B are not met, the ability to isolate the secondary containment and start the SGT System cannot be ensured. Therefore, further actions must be performed to ensure the ability to maintain the secondary containment function. Isolating the associated zone (closing the ventilation supply and exhaust automatic isolation dampers) and starting the associated SGT subsystem (Required Actions C.1.1 and C.2.1) performs the intended function of the instrumentation and allows operation to continue.

Alternately, declaring the associated SCIVs or SGT subsystem(s) inoperable (Required Actions C.1.2 and C.2.2) is also acceptable since the Required Actions of the

(continued)

REV. 2

P63

INSERT B3.3-216A

The Surveillances are modified by a third Note (Note 3) to indicate that for Functions 3 and 4, a CHANNEL CALIBRATION when a channel is placed in an inoperable status solely for performance of required testing or maintenance, entry into associated Conditions and Required Actions may be delayed for up to 6 hours for a CHANNEL FUNCTIONAL TEST and for up to 24 hours for a CHANNEL CALIBRATION or maintenance, provided the downscale trip of the inoperable channel is placed in the tripped condition. Upon completion of the Surveillance or maintenance, or expiration of the 6 hour or 24 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken.

B 3.3 INSTRUMENTATION

B 3.3.8.1 Loss of Power (LOP) Instrumentation

BASES (P11) <entire page>

BACKGROUND

Successful operation of the required safety functions of the Emergency Core Cooling Systems (ECCS) is dependent upon the availability of adequate power sources for energizing the various components such as pump motors, motor operated valves, and the associated control components. The LOP instrumentation monitors the 4.16 kV ~~emergency buses~~ shutdown boards. Offsite power is the preferred source of power for the 4.16 kV ~~emergency buses~~ shutdown boards. If the monitors determine that insufficient power is available, the ~~buses~~ boards are disconnected from the offsite power sources and connected to the onsite diesel generator (DG) power sources.

Each 4.16 kV ~~emergency bus~~ shutdown board has its own independent LOP instrumentation and associated trip logic. The voltage for each ~~bus~~ board is monitored at two levels, which can be considered as two different undervoltage Functions: Loss of Voltage and 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage Degraded Voltage. Each Function causes various ~~bus~~ board transfers and disconnects. ~~Each~~ Function is monitored by ~~two~~ three undervoltage relay channels for each ~~emergency bus~~ shutdown board, whose outputs are arranged in a two-out-of-~~two~~ three logic configuration (Ref. 1). The channels ~~include electronic equipment (e.g., trip units) that~~ compare measured input signals with pre-established setpoints. When the setpoint is exceeded, ~~the channel output relay actuates, which then outputs a~~ LOP trip signal to the ~~trip~~ logic. ~~deenergizes,~~

INSERT B3.3-220

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

The LOP instrumentation is required for Engineered Safety Features to function in any accident with a loss of offsite power. The required channels of LOP instrumentation ensure that the ECCS and other assumed systems powered from the DGs, provide plant protection in the event of any of the Reference 2, 3, and 4 analyzed accidents in which a loss of offsite power is assumed. The initiation of the DGs on loss of offsite power, and subsequent initiation of the ECCS, ensure that the fuel peak cladding temperature remains below the limits of 10 CFR 50.46.

for two-of-three degraded voltage channels, the logic energizes timers which provide a

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(continued)

BASES (P11) (except as marked)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The specific Applicable Safety Analyses, LCO, and Applicability discussions are listed below on a Function by Function basis. The channel devices for each shutdown board are listed in Table B3.3.8.1-1. (P11)

1. 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage (Loss of Voltage)

Loss of voltage on a 4.16 kV ~~emergency bus~~ shutdown board indicates that offsite power may be completely lost to the respective ~~emergency bus~~ shutdown board and is unable to supply sufficient power for proper operation of the applicable equipment. Therefore, the power supply to the ~~bus~~ board is transferred from offsite power to DG power ~~when the voltage on the bus drops below the Loss of Voltage Function Allowable Values (loss of voltage with a short time delay)~~. This ensures that adequate power will be available to the required equipment. INSERT B3.3-222A

~~The Bus Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that power is available to the required equipment.~~ The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that power is available to the required equipment.

~~Two~~ One channel~~s~~ of 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage shutdown board (Loss of Voltage) Function per associated ~~emergency bus~~ shutdown board ~~are~~ is ~~only~~ required to be OPERABLE when the associated DG is required to be OPERABLE to ensure that no single instrument failure can preclude the DG function. ~~(Two channels input to each of the three DGs.)~~ Refer to LCO 3.8.1, "AC Sources—Operating," and 3.8.2, "AC Sources—Shutdown," for Applicability Bases for the DGs. stet

2. 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage (Degraded Voltage)

A reduced voltage condition on a 4.16 kV ~~emergency bus~~ shutdown board indicates that, while offsite power may not be completely lost to the respective ~~emergency bus~~ shutdown board, available power may be insufficient for starting large ECCS motors without risking damage to the motors that could disable the ECCS function. Therefore, power supply to the ~~bus~~ board is transferred from offsite power to onsite DG power when the voltage on the ~~bus~~ board drops below the Degraded Voltage Function Allowable Values

(continued)

P11 (except as marked)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

2. 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage (Degraded Voltage) (continued)

(degraded voltage with a time delay). This ensures that adequate power will be available to the required equipment.

The ~~Bus~~ Board Undervoltage Allowable Values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The Time Delay Allowable Values are long enough to provide time for the offsite power supply to recover to normal voltages, but short enough to ensure that sufficient power is available to the required equipment.

Three ~~Two~~ ~~One~~ channels of 4.16 kV ~~Emergency Bus~~ Shutdown Board Undervoltage (Degraded Voltage) Function per associated ~~bus~~ board ~~are, only~~ required to be OPERABLE when the associated DG is required to be OPERABLE is to ensure that no single instrument failure can preclude the DG function. ~~(Two channels input to each of the three emergency buses and DGs.)~~ Refer to LCO 3.8.1 and LCO 3.8.2 for Applicability Bases for the DGs.

ACTIONS.

A Note has been provided to modify the ACTIONS related to LOP instrumentation channels. Section 1.3, Completion Times, specifies that once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies that Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable LOP instrumentation channels provide appropriate compensatory measures for separate inoperable channels. As such, a Note has been provided that allows separate Condition entry for each inoperable LOP instrumentation channel.

P51

INSERT B 3.3-223A

P51

A.1

~~With one or more channels of a Function inoperable, the Function is not capable of performing the intended function. Therefore, only 1 hour is allowed to restore the inoperable~~

(continued)

INSERT B3.3-223A (Page 1 of 2)

P51

A.1 and A.2

Insert B3.3-223A (Page 1a of 2) Item A

With one of the ~~three~~ ~~phase-to-phase~~ degraded voltage relay channels inoperable, Required Action A.~~X~~2 provides a 15 day allowable out of service time to restore the relay to OPERABLE status. The 15 day allowable out of service time is justified based on the two-out-of-three permissive logic scheme channel provided for these relays. If the inoperable relay channel cannot be restored to OPERABLE status within the allowable out of service time, the degraded voltage relay channel must be placed in the tripped condition per Required Action A.~~X~~2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

B.1 and B.2

Insert B3.3-223A (Page 1a of 2) - Item B

With one or more loss of voltage relay channels inoperable, the Function is not capable of performing the intended function. Required Action B.~~X~~2 provides a 10 day allowable out of service time ~~since~~ the degraded voltage relay channels on the same shutdown board ~~is~~ are independent of the loss of voltage relay channels and will continue to function and start the diesel generators on a complete loss of voltage ~~and associate timers~~. If the inoperable channel(s) cannot be restored to OPERABLE status within the allowable out of service time, the channel(s) must be placed in the tripped condition per Required Action B.~~X~~2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

INSERT B3.3-223A (Page 1a of 2) - Item A

... provided the other two ~~phase-to-phase~~ degraded voltage relay channels and the loss of voltage relay channels on that shutdown board are OPERABLE. Immediate verification of the OPERABILITY of the other ~~phase-to-phase~~ degraded voltage relay channels and loss of voltage relay channels are therefore required (Required Action A.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition C or D, as applicable, must be entered immediately.

INSERT B3.3-223A (Page 1a of 2) - Item B

... provided ~~the~~ two or more degraded voltage relay channels and associated timers on that shutdown board ~~is~~ are OPERABLE. Immediate verification of the OPERABILITY of ~~the~~ two or more degraded voltage relay channels and associated timers ~~is~~ are therefore required (Required Action B.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition D must be entered immediately. The 10 day allowable out of service time is justified since ...

(P51)

INSERT B3.3-223A (Page 2 of 2)

C.1 and C.2

(or one or more timers) — Insert B3.3-223A (Page 2a of 2)

With ~~one~~ two or more degraded voltage relay channels inoperable, the Function is not capable of performing the intended function. Required Action C.1 2 provides a 10 day allowable out of service time, ~~since~~ the loss of voltage relay channels on the same shutdown board ~~is~~ are independent of the degraded voltage relay channels and will continue to function and start the diesel generators on a complete loss of voltage. If the inoperable channels cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action C.1 2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

D.1 and D.2

With ~~the~~ two or more degraded voltage relay channels (or one or more associated timers) and the loss of voltage relay channel(s) inoperable on the same shutdown board, the associated diesel generator will not automatically start upon degraded voltage or complete loss of voltage on that shutdown board. In this situation, Required Action D.2 provides a 5 day allowable out of service time provided the other shutdown boards and undervoltage relay channels ~~is~~ are OPERABLE. Immediate verification of the OPERABILITY of the other shutdown boards and undervoltage relay channels ~~is~~ are therefore required (Required Action D.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition E must be entered immediately. The 5 day allowable out of service time is justified based on the remaining redundancy of the 4.16 kV Shutdown Boards. The 4.16 kV Shutdown Boards have a similar allowable out of service time. If the inoperable channel cannot be restored to OPERABLE status within the allowable out of service time, the channel must be placed in the tripped condition per Required Action D.2. Placing the inoperable channel in trip would conservatively compensate for the inoperability, restore capability to accommodate a single failure (within the LOP instrumentation), and allow operation to continue. Alternately, if it is not desired to place the channel in trip (e.g., as in the case where placing the channel in trip would result in a DG initiation), Condition E must be entered and its Required Action taken.

P51

INSERT B3.3-223A (Page 2a of 2)

... provided the loss of voltage relay channel~~s~~ on that shutdown board ~~is~~ are OPERABLE. Immediate verification of the OPERABILITY of the loss of voltage relay channel~~s~~ ~~is~~ are therefore required (Required Action C.1). This may be performed as an administrative check by examining logs or other information to determine if this equipment is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate OPERABILITY of this equipment. If the OPERABILITY of this equipment cannot be verified, however, Condition D must be entered immediately. The 10 day allowable out of service time is justified since ...

BASES

BACKGROUND (continued)

contactors ~~circuit breakers~~ (P11) has an associated independent set of Class 1E overvoltage, undervoltage, and underfrequency sensing logic. Together, a ~~circuit breaker~~ contactor and its sensing logic constitute an electric power monitoring assembly. If the output of the MG set exceeds predetermined limits of overvoltage, undervoltage, or underfrequency, for > four seconds, a ~~trip coil~~ trip relay driven by this logic circuitry opens the ~~circuit breaker~~ contactor, which removes the associated power supply from service. (P11)

(P11) The timer is common to the three trip relays

APPLICABLE SAFETY ANALYSES

The RPS electric power monitoring is necessary to meet the assumptions of the safety analyses by ensuring that the equipment powered from the RPS buses can perform its intended function. RPS electric power monitoring provides protection to the RPS and other systems that receive power from the RPS buses, by acting to disconnect the RPS from the power supply under specified conditions that could damage the RPS bus powered equipment.

RPS electric power monitoring satisfies Criterion 3 of the NRC Policy Statement. (Ref 2) (P8)

LCO

The OPERABILITY of each RPS electric power monitoring assembly is dependent on the OPERABILITY of the overvoltage, undervoltage, and underfrequency logic, as well as the OPERABILITY of the associated ~~circuit breaker~~ contactor. (P11) Two electric power monitoring assemblies are required to be OPERABLE for each inservice power supply. This provides redundant protection against any abnormal voltage or frequency conditions to ensure that no single RPS electric power monitoring assembly failure can preclude the function of RPS bus powered components. Each inservice electric power monitoring assembly's trip logic setpoints are required to be within the specified Allowable Value. The actual setpoint is calibrated consistent with applicable setpoint ~~methodology assumptions~~. procedures (nominal trip setpoint) (P11)

Allowable Values are specified for each RPS electric power monitoring assembly trip logic (refer to SR 3.3.8.2.2). Nominal trip setpoints are specified in the setpoint calculations. The nominal setpoints are selected based on engineering judgment and operational experience to ensure that the setpoints do not exceed the Allowable Value between CHANNEL CALIBRATIONS. Operation with a trip setpoint less (P45)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.8.2.2 (continued)

P5 P43

The Frequency is based on the assumption of a 184 day ~~18 month~~ calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

SR 3.3.8.2.3

Performance of a system functional test demonstrates that, with a required system actuation (simulated or actual) signal, the logic of the system will automatically trip open the associated power monitoring assembly. Only one signal per power monitoring assembly is required to be tested. This Surveillance overlaps with the CHANNEL CALIBRATION to provide complete testing of the safety function. The system functional test of the Class 1E ~~circuit breakers~~ contactors, is included as part of this test to provide complete testing of the safety function. If the ~~breakers~~ contactors are incapable of operating, the associated electric power monitoring assembly would be inoperable. P11

The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

B1 1. FSAR, Section ~~[8.3.1.1.4.B]~~ 7.2.3.2.

~~2. NRC Generic Letter 91-09, "Modification of Surveillance Interval for the Electrical Protective Assemblies in Power Supplies for the Reactor Protection System~~ P11

P8 23 NRC No. 93-102, "Final Policy Statement on Technical Specification Improvements," July 23, 1993.

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

NUREG-1433 BWR/4 STANDARD TECHNICAL SPECIFICATIONS/BASES JUSTIFICATION FOR CHANGES

Replaced pages 1 through 18 Revision 1 with pages 1 through 16 Revision 2

JUSTIFICATION FOR CHANGES TO NUREG-1433
SECTION 3.3 - INSTRUMENTATION

BRACKETED PLANT SPECIFIC INFORMATION

B1 Brackets removed and optional wording preferences revised as necessary to reflect appropriate plant specific requirements.

B2 Brackets removed and values revised as necessary to reflect plant specific design.

B3 Bracketed requirements removed and optional wording deleted since it is not applicable to BFN plant design. This includes deletion of Reviewer Notes since they are provided as a reviewer aid and are not applicable to plant specific Improved Standard Technical Specifications.

NON-BRACKETED PLANT SPECIFIC CHANGES

P1 The specifications (Specifications, LCOs, Conditions, Required Actions, and SRs) have been renumbered to reflect the deletion or addition of requirements by other comment numbers, or changes to the SR Frequencies. Bases changes were made consistent with the specifications (e.g., a change due to deletion of a bracketed Specification requirement).

P2 This SR has not been included in the BFN ISTS. This SR was added to the NUREG when the new GE Analog Transmitter Trip system(s) (ATTS) was licensed and installed in BWRs. At BFN, the setpoint and scaling calculations for instrument loops with analog trip units assume the trip units are calibrated at the same frequency as the CHANNEL CALIBRATION. Therefore, a separate CHANNEL CALIBRATION for the trip unit is not required. Drift between calibration intervals is accounted for in BFN calculations. Appropriate Bases changes have been made.

P3 The specific value will be relocated to the COLR. This will help eliminate the number of Technical Specification changes required due to these values changing. The COLR is a controlled document in which any changes are made in accordance with 10 CFR 50.59.

P4 BFN Technical Specifications currently do not require response time testing. Industry generic studies show that response time changes (times getting longer), that could impact safety, do not normally vary such that they would not be detected during other required surveillances (e.g., CHANNEL CALIBRATIONS). Since the addition of this SR is a major burden to Browns Ferry, with little gain in safety, the SR has not been added.

P5 Typographical/grammatical error corrected.

P6 BFN is not licensed for single loop operation; therefore, this note does not apply.

P7 This change has been made for consistency with other occurrences in the TS and Bases.

P8 The proper reference has been provided.

P9 Based on the BFN instrumentation logic design, the appropriate instrumentation logic, system operation and design description has been provided. At Brown's Ferry, secondary containment and RWCU isolate on Reactor Vessel Water Level-Low, Level 3.

P10 This table has been deleted since it provides generic rather than plant specific information. The information in the Table could be misleading as to which plant-specific analyses take credit for these channels to perform a function during accident and transient scenarios.

P11 Changes were made to provide additional information or clarity, to use plant specific terminology, to describe plant specific design requirements, or to depict plant specific instrumentation requirements.

P12 This clarification has been added since the RPS is also required to be OPERABLE during conditions that are not MODES.

P13 As stated in the Reviewer Note at the bottom of the page, this Table has been modified to reflect the Browns Ferry Regulatory Guide (RG) 1.97 requirements as specified in BFN General Design Criteria No. BFN-50-7307, Revision 4, "Post-Accident Monitoring," dated June 22, 1993. The Table includes all RG 1.97 Type A and Non-Type A, Category 1 instruments specified in the Browns Ferry RG 1.97 Safety Evaluation Report (SER) dated June 23, 1988. The Reviewer Note has been deleted since it is not applicable in a plant specific ISTS. Appropriate Bases changes have been made to properly describe the Functions required by Browns Ferry.

P14 At BFN Functions 3.b, 3.c, 4.b and 4.c have only one trip system. Although four channels are available and normally operable, only three are required per the BFN current licensing basis (CLB). BFN chooses to maintain this CLB requirement. The Bases have been revised to clarify that each function is considered to have only one trip system since the output from the logic trips a common relay that initiates the isolations.

P15 Not Used.

P16 As currently written the BASES for EOC-RPT Instrumentation in NUREG-1433 would require placing all of the Turbine Stop Valve-Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low trip systems in inoperable status any time a Turbine Bypass Valve is opened. BFN's Current Technical Specifications do not contain this requirement and chooses not to implement this part of the NUREG. While normal operation above 30% RTP would not require the Turbine Bypass valves to be open, each valve is required to be opened at least once per 31 days to comply with ITS SR 3.7.5.1 to "Verify one complete cycle of each main turbine bypass valve." The 30% RTP signal is a bypass for the functions of concern and unless bypass occurs do not affect the operability of the functions. The 30% RTP parameter is derived from the first stage turbine pressure and thus opening the bypass valve may potentially cause the bypass to occur but it is more likely that it will not if operating at higher RTP (and thus higher pressure). Since opening a bypass valve will not necessarily result in bypassing the Turbine stop Valve - Closure and Turbine Control Valve Fast Closure, Trip Oil Pressure - Low functions declaring them inoperable is unnecessary unless bypass actually occurs. If 30% RTP bypass occurs, it is alarmed in the Main Control Room. Bypassing the functions while a bypass valve is open would result in declaring them inoperable. Based on the above, the following deviations from NUREG-1433 are proposed to be made to the ITS BASES:

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY, Turbine Stop Valve - Closure will delete "to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER $\geq$ 30% RTP" and replaced with "opening the turbine bypass valves may affect this function, therefore to consider this function OPERABLE bypass of the function must not occur when bypass valves are open."

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY, Turbine Control Valve Fast Closure, Trip Oil Pressure - Low will delete "to consider this Function OPERABLE, the turbine bypass valves must remain shut at THERMAL POWER $\geq$ 30% RTP" and replaced with "opening the turbine bypass valves may affect this function, therefore to consider this function OPERABLE bypass of the function must not occur when bypass valves are open."

SURVEILLANCE REQUIREMENTS, SR 3.3.4.1.5 (renumbered to 3.3.4.1.2 in ITS) will delete "Because main turbine bypass flow can affect this setpoint nonconservatively (THERMAL POWER is derived from first stage pressure) the main turbine bypass valves must remain closed at THERMAL POWER $\geq$ 30% RTP to assure calibration remains valid."

The deviations will meet the intent of not allowing opening the bypass valves to impact the bypass of the trip systems such that it occurs when RTP $\geq$ 30% without declaring the trip systems inoperable. Based on the above, it is concluded that this is an acceptable change.

P17 The words or frequencies have been modified to correspond to the appropriate LCO or SR.

P18 These changes reflect the BFN specific design, analyses, licensing basis, and/or nomenclature.

P19 Deleted.

P20 Browns Ferry Nuclear Plant is not licensed with the option to have a lower count rate. Therefore, this option has not been used in the BFN ISTS.

P21 The RWM is not a hardwired system; therefore, "hardwired" has been removed from the Bases description.

P22 This change has been made for consistency with the manner in which LCOs are to be referenced.

P23 LCO 3.3.6.1, Function 5.h and 6.b, LCO 3.3.6.2, Function 1 and LCO 3.3.7.1, Function 1, Reactor Vessel Water Level - Low, Level 3, and LCO 3.3.6.1, Function 6.c, LCO 3.3.7.1, Function 2, Drywell Pressure - High, are common to the RPS and may only be extended to 12 hours rather than 24 based on NEDC-30851-P-A, Supplement 2, March 1989, NEDC-31677-P-A, July 1990 and GENE-770-06-1, February 1991.

P24 BFN does not have a RBM bypass time delay, therefore, this function has been deleted.

P25 Currently, the BFN Technical Specifications (TS) allow the main steam line (MSL) temperature switches to be bypassed for 4 hours when normal ventilation is not available (e.g., during the performance of secondary containment leak rate tests) to avoid a MSL isolation transient. This provision was added to BFN TS by Amendment Nos. 110, 103 and 76 for Units 1, 2, and 3 respectively. The NRC Safety Evaluation concluded that the change was acceptable based on the short inoperability interval and the interim compensatory measures concerning monitoring of space temperatures by the operators. BFN chooses to maintain these provisions and has added it to proposed Action B (second Completion Time - applies only to this function when normal ventilation is not available). Specific additional compensatory measures that are currently required when MSL temperature channels are inoperable due to this intentional action were added (operator to monitor control room indications of the

affected space temperatures and promptly close the MSL isolation valves in the event of rapid increases in temperature) to the Bases. Additional information was provided in the Bases for LCO 3.3.6.1, Condition B to clarify the acceptability of intentionally entering Condition B to avoid a MSL isolation transient when recovering from a loss of ventilation transient.

P26 In the NUREG, the Applicable Safety Analysis, LCO and Applicability discussion for Function 1.a (IRM Neutron Flux-High) states that in MODE 1, the APRM System and the RWM provide protection against control rod withdrawal error events. The RBM is used to provide protection against control rod withdrawal error events, not the RWM. The RWM enforces specific control rod sequences designed to mitigate the consequences of a control rod drop accident during low power operations.

P27 This instrumentation does not provide a "level 8" signal, therefore, references to "level 8" have been deleted.

P28 These sentences have been deleted since they are providing reviewer's information and are not applicable in a plant specific ISTS.

P29 The Table list of the Remote Shutdown System instrumentation and controls is proposed to be removed from the Technical Specifications and relocated to the TS Bases, a plant controlled document. This change is consistent with the provisions of Generic Letter 91-08 for the removal of lists and has been recently approved for Clinton Power Station (Amendment No. 68) on that basis. Appropriate Bases changes were made.

P30 SR 3.3.3.2.1 in NUREG-1433 requires performance of a CHANNEL CHECK for each required instrumentation channel, that is normally energized, every 31 days. There are no CTS applicable to Backup Control for BFN. The current licensing bases for BFN for Backup Control Inspection and Test is located in FSAR Section 7.18.6. This section states "Operability of components from the Backup Control Center will be tested to the extent practical once per operating cycle. This includes testing of transfer of control of active components and instrument calibration." Since there are no current requirements to perform monthly CHANNEL CHECKS, BFN chooses not to incorporate NUREG-1433, SR 3.3.3.2.1 into the ITS. Necessary changes to the BASES and administrative changes for SR numbering will be made.

P31 The proper criterion from the Final Policy Statement has been used. The current wording was developed prior to the issuance of the Final Policy Statement, which uses Criterion 4 in place of the NUREG words.

P32 Browns Ferry has only one required division; thus, this parenthetical phrase has been deleted.

P33 The load reject and turbine trip are the only two events referenced in the FSAR; thus, any reference to other events has been deleted.

P34 This information has been deleted since it is already found in the Background section where all other logic descriptions are located.

P35 The Manual Initiation Channel requirements have been deleted since they are not currently required at Browns Ferry. Considering the logic design, there is not a single switch that will start all subsystems of the associated ECCS system, ADS, or RCIC, or close all the associated PCIVs and SCIVs. Since the Note for Required Action G.1 of LCO 3.3.5.1 was written assuming Manual Initiation channels for ADS were in the TS, it is not needed and has been deleted. Note 2 for the Surveillances in LCOs 3.3.5.1 and 3.3.5.2 have also been modified to reflect these deletions. ACTION G of LCO 3.3.6.1 and any reference to it has been deleted since it applies only to Manual Initiation channels.

P36 Browns Ferry accident analyses for recirculation line breaks do not take credit for HPCI being operable. Therefore, discussion of this function as it relates to HPCI initiation has been changed appropriately.

P37 This proposed change adds Function 1.e, Core Spray Pump Start-Time Delay Relay, to Table 3.3.5.1-1, "ECCS Instrumentation." This Function is specific to the BFN design.

P38 RCIC has no automatic transfer from CST to the suppression chamber. Operators are directed to transfer RCIC when HPCI auto transfers on low CST level or high suppression chamber level and before RCIC trips on low suction pressure (15" Hg vacuum) or when CST level indicates $\leq$ 11 feet.

P39 The reactor water cleanup (RWCU) system differential flow, condenser vacuum and drywell high radiation isolation instrumentation are not included in the BFN ISTS since they are not considered primary containment isolation system instruments at BFN and are not a requirement of current Technical Specifications. Appropriate Bases changes have been made.

P40 This function is not included in the BFN design and has been deleted.

P41 A Logic System Functional Test with a 184 day Frequency has been added to address plant specific requirements for testing the Control Room Air Supply Duct Radiation Monitor (SR 3.3.7.1.4).

P42 A Channel Calibration with a 122 day Frequency has been added to address plant specific requirements for testing the RWCU area temperature functions.

P43 The appropriate calibration frequency has been called out based on the instruments capability.

P44 Deleted.

P45 The correct reason for the Allowable Value or trip setpoints has been provided.

P46 Current Technical Specifications (CTS) allow one channel of the Control Room Air Supply Duct - High function to be inoperable provided that repairs are initiated and the other channel is functionally tested every 24 hours. CTS also allows both channels to be inoperable for 30 days provided that the alternate monitoring capability is functionally tested and the operator is administratively required to start CREVS upon receipt of an alarm from the alternate monitors. BFN chooses to maintain these current licensing basis provisions. Proposed Action D has been created to include these provisions. Action C has been revised to reflect appropriate actions and allowable out of service times (i.e., 24 hours instead of 6 hours) for the Reactor Zone Exhaust Radiation-High and Refueling Floor Exhaust Radiation-High functions.

P47 At BFN, CREVS is initiated by the following functions: Reactor Vessel Water Level-Low, Level 3; Drywell Pressure-High, Reactor Zone Exhaust Radiation-High, Refueling Floor Exhaust Radiation-High, and Control Room Air Supply Duct Radiation-High. As such, the Main Steam Line Flow-High function has been deleted and the Reactor Zone Exhaust Radiation-High function has been added.

P48 This Note has been deleted since Browns Ferry design does not have a toxic gas protection mode. Appropriate Bases changes have been made.

P49 BFN currently performs channel checks once every 24 hours. This frequency is adequate to ensure gross failure of instrumentation has not occurred. BFN operating experience has demonstrated that the 24 hour channel check frequency is adequate to ensure that instrumentation operates properly between calibrations.

P50 SR 3.3.4.2.1 has been modified to apply only to the Reactor Vessel Water Level-Low Function. BFN CTS has no requirements to perform Channel Check on either the Reactor Vessel Water Level Low or the Reactor Vessel Dome Pressure High ATWS-RPT instrumentation as evidence by Table 4.2.L "Instrument Check" column being marked N/A for both functions. BFN chose to implement the NUREG-1433 requirement for a Channel Check for the Reactor Vessel Water Level Low function because the instrumentation already had a Channel Check requirement in other CTS sections as addressed by JFC A5. BFN chooses not to implement the NUREG-1433 requirement for a Channel Check for the Reactor Vessel Dome Pressure

High because the perceived benefit beyond the normal observation of the parameter is small and does not justify the additional requirements.

P51 NUREG-1433, LCO 3.3.8.1, ACTION A was deleted and proposed BFN ISTS ACTIONS A, B, C, and D were added to retain BFN current Technical Specification allowable outage times for undervoltage relays. The degraded voltage sensing relays provide a start signal to the diesel generators (DG) in the event that a deteriorated voltage condition exists on a 4.16 kV shutdown board. This starting signal is independent of the starting signal generated by the complete loss of voltage relays and will continue to function and start the DGs on complete loss of voltage should the loss of voltage relays become inoperable. The 15 day inoperable time limit specified when one of the three phase-to-phase degraded voltage relays is inoperable is justified based on the two-out-of-three logic scheme provided with these relays.

P52 This discussion about analytical limits and the derivation of the Allowable Values and trip setpoints has been deleted since it does not apply to this Function.

P53 The Applicability and Condition D of LCO 3.3.8.2 have been revised consistent with the BFN design and licensing basis. At BFN the logic for the RHR Shutdown Cooling isolation valves is not powered by RPS.

P54 BFN safety relief valves do not have a low-low set function. Deleted LCO and LCO Bases.

P55 Notes 1 and 2 to NUREG SRs 3.3.1.1.11 and 3.3.1.1.13 have been relocated to the appropriate SR (proposed BFN SR 3.3.1.1.9).

P56 A 92 day Channel Calibration (SR 3.3.3.1.2) was added for the Drywell & Torus H_2 Analyzer to maintain consistency with the current BFN licensing basis and the requirements of NUREG-1433. A 184 day Channel Calibration (SR 3.3.3.1.3) was added for the Reactor Pressure Indication instruments in accordance with BFN plant specific analysis that identify the required calibration frequency.

P57 This change deletes the Note to Specification 3.3.3.1 SRs. The Note states that the SRs apply to each Function in the Table. This Note is no longer required because each SR has been modified to direct which Function it applies to, since the SRs do not apply to all the Functions in Table 3.3.3.1-1.

P58 BFN current licensing basis does not require verification of the APRM Flow Biased Simulated Thermal Power - High time constant (NUREG SR 3.3.1.1.14). The gain in safety by the addition of the SR does not justify the cost associated with the added surveillance. The

calibration of the APRM Flow instrumentation is already very time intensive and must be performed during a outage. Any significant changes in one of the timing circuits (which are normally stable once burned in) would result in a observable delta between indicated values. BFN Current Technical Specifications BASES Section 2.1.a.1 "APRM Flow-Biased High Flux Scram Trip Setting (RUN Mode)" states "No safety credit is taken for flow-biased scrams." Since this SR would provide minimal gain in the APRM Flow Biased Simulated Thermal Power accuracy or reliability and no safety credit is taken for the function it is acceptable to delete this SR.

P59 Current BFN Technical Specifications do not require channel checks be performed on the following functions: Table 3.3.1.1, Function 6; Table 3.3.5.1-1, Functions 1.b, 1.c, 2.b, 2.c, 2.d, 3.b, 4.b, 5.b; Table 3.3.6.1-1 Function 2.b, 5.a through 5.f; Table 3.3.6.2-1, Function 2; and Table 3.3.7.1, Function 2. As such, BFN chooses not include a channel check requirement in the proposed BFN ISTS for these functions.

P60 BFN analyses take credit for an RPS trip on low scram pilot air header pressure. Therefore, this function along with appropriate actions and surveillance requirements have been added to the proposed Specification and associated Bases. This SR is only applicable to Units 2 and 3 because the low scram pilot air header pressure trip scram function has not been incorporated into the current Unit 1 Technical Specifications because the installation has not been completed. The Unit 1 Technical Specifications will be updated to include this SR prior to placing the scram pilot air header pressure trip function in service.

P61 Not used.

P62 BFN is not licensed for single loop operation. Accordingly, LCO 3.3.4.1, Required Action C.1 and LCO 3.3.4.2, Required Action D.1 and their corresponding Bases discussion have been deleted.

P63 BFN will maintain the current licensing basis provision of Note 11 to Table 3.2.A which allows one channel of the Reactor Zone and Refueling Zone Exhaust Radiation System to be inoperable for up to 4 hours for functional testing and for up to 24 hours for calibration and maintenance as long as the downscale trip of the inoperable channel is placed in the tripped condition. BFN has extended the 4 hour period to 6 hours based on reliability analyses (NEDC-30851-P-A, NEDC-31677-P-A and GENE-770-06-1). Refer to Justification LB1 of CTS Markup for Specifications 3.3.6.2, and 3.3.7.1.

P64 BFN current licensing basis allows 1 of 16 MSL temperature switches to be bypassed for 30 days. This was approved by Amendment Nos. 107, 101, and 74 (NRC Safety Evaluation dated August 9, 1994) for Units 1, 2 and 3

respectively. The Safety Evaluation concluded that power operation for 30 days with 15 of 16 Operable is an acceptable risk. Required Action A.2, which includes a 30 day Completion Time, and the Notes to Required Actions A.1 and A.2, have been added to retain this current licensing basis provision.

P65 At BFN, Functions 3.d and 3.e have two channels available but only require one OPERABLE since HPCI is not required to be single failure proof. This current licensing basis requirement will be retained.

P66 A Channel Calibration with a 184 day Frequency has been added to address plant specific requirements for testing the Reactor Steam Dome Pressure - Low and Suppression Pool Water Level functions.

P67 For NUREG 3.3.7.1 Required Actions B.1 and C.1 the words "in both trip systems" has been deleted since these words imply each trip system is capable of feature initiation which is not the case for all CREV System Instrumentation Functions for BFN. Several channels used for CREV initiation, Secondary Containment Isolation, and Primary Containment Isolation are common. Some of these channels input to the PCIS trip systems which provide signals to logic systems that initiate the CREV System, Secondary Containment Isolation, and Primary Containment Isolation. Logic arrangement in these cases are such that one trip channel in both PCIS trip systems must trip to cause feature initiation. The deletion of these words will not result in a change in the intent of these Required Actions and the proposed ISTS Bases for these required actions define the requirements for feature initiation capability to be maintained. The proposed ISTS Bases has also been changed to address the above.

P68 None of the Loss of Offsite Power (LOP) instruments currently require a channel check or channel functional test. BFN proposed Surveillance Requirements, which do not include NUREG-1433 SRs 3.3.8.1.1 and 3.3.8.1.2, are based on current licensing basis.

P69 At BFN, one CREV unit can supply all three control rooms. One CREV subsystem automatically starts upon receipt of an initiation signal and the other subsystem remains in standby unless the subsystem selected to automatically start fails to start. If the selected subsystem fails to start, the standby subsystem will start after a time delay. There are two control logic systems which start their associated CREV subsystem. Each control logic system receives inputs from the CREV System Instrumentation Functions by means of the PCIS trip systems Reactor Zone and Refueling Zone Exhaust Radiation Monitor divisional trip systems, and the Control Room Air Supply Duct Radiation Monitor trip systems.

P70 A Channel Calibration with a 92 day Frequency has been added to address plant specific requirements for testing the Control Room Air Supply Duct Radiation - High Function.

P71 Deleted.

P72 The applicability for LCO 3.3.8.2 has been clarified to state withdrawal of a control rod in MODE 4 is only allowed by Special Operations LCO 3.10.4.

P73 NUREG B 3.3.1.2, Required Action C.1, has been revised to delete the words "from full power conditions" since this action is taken from MODE 2. This change is consistent with the wording of NUREG-1434, Revision 1, BWR/6 Standard Technical Specifications.

P74 The channel check requirement for this Function has been deleted since the device has no indication that can be checked.

P75 The first sentence of the last paragraph on page B 3.3-92 should not indicate the basis for ATWS is "to protect against common mode failures" of the RPS system. The system has been designed and analyzed to ensure there are no common mode failures for RPS. The ATWS modifications were implemented as a special event mitigation system for catastrophic/multiple failures of the RPS which is beyond the design/licensing basis of BFN. The term "common mode failure" has been changed to "catastrophic/multiple failures" in the BASES for 3.3.4.2.

P76 Proposed footnote (e) has been added to Table 3.3.5.1-1 to define the required number of channels operable per RHR (LPCI) pump.

P77 The discussion related to single failure protection has been deleted since the HPCI/RCIC systems are not required to be single failure proof.

P78 At BFN, the Allowable Value for the Reactor Vessel Water Level - Low, Level 3 (Confirmatory) Function is not the same as the Allowable Value for the Reactor Protection System Reactor Vessel Water Level - Low, Level 3 Function. Therefore, this statement has been deleted.

P79 The last sentence of the first paragraph of NUREG SR 3.3.1.1.6 has been deleted since it implies that calibration of 1st stage turbine pressure devices is not to be performed with bypass valves open when ≥ 30% RTP. The calibration of these devices (pressure transmitters) would not be affected by the bypass valves opening.

P80 Note 3 has been added to the Actions Table for Specification 3.3.3.1 to clarify that for Function 6, separate Condition entry applies to each penetration flow path. The Bases have been modified to reflect the

addition of this note. Without the clarifying note it could be construed that condition C, "One or more Functions with two required channels inoperable" would be invoked when indication of two valves in separate penetrations were inoperable. The number (82) of Primary Containment Isolation Valves (PCIVs) that the function pertains to is large enough that more than one indication may be inoperable at one time. Many of the valves are located inside primary containment which would make it difficult to "Restore one required channel to OPERABLE status" within seven days which would result in the invocation of Conditions E and F which would result in being in MODE 3 within 12 hours. This ACTION is excessive because one valve per penetration would still have position indication available to confirm isolation if necessary. In addition, should three PCIVs lose position indication then LCO 3.0.3 would be invoked because "an associated ACTION is not provided" for more than two required channels inoperable. Invocation of LCO 3.0.3 would result in starting shutdown within one hour which is excessive. Function 6, PCIV position indication, is necessary to verify containment isolation of a penetration. Each penetration flow path that contains two active valves would still have at least one valve position indication available if two penetrations each lose one valve position indication. Thus, the capability to verify isolation of a penetration would be maintained. Based on the preceding information it justified to consider each penetration separately for entry into LCO 3.3.3.1.

P81 Required Actions D.2.1 and D.2.2 of Specification 3.3.5.1 have been deleted since BFN only requires one channel to be operable. Loss of this channel results in loss of initiation capability; therefore, the 24 hour allowable outage time cannot be applied at BFN. BFN will maintain the CTS requirement of declaring HPCI inoperable.

P82 Function 2.g, LPCI Pump Discharge Flow-Low (Bypass) of NUREG Table 3.3.5.1-1, has not been included in the proposed BFN ISTS since LPCI flow rates assumed in the BFN LOCA analyses can be achieved with the minimum flow valve in the open position. The only purpose of this function is to provide equipment protection (open minimum flow return line valve to preclude pump overheating). As such, it has not been included in the proposed BFN ISTS.

P83 Deleted.

P84 Deleted.

P85 Reference to the CS test line isolation valve as a primary containment isolation valve (PCIV) in the Background for Specification 3.3.5.1 Bases has been deleted. At BFN, this valve is not a PCIV.

P86 The Applicable Safety Analyses, LCO, and Applicability discussion in the Bases for Specification 3.3.4.2 (page B 3.3-92) of the NUREG states that ATWS-RPT instrumentation trip setpoints consider severe environment errors (for channels that must function in harsh environments as defined by 10 CFR 50.49). ATWS-RPT instruments are not assumed to function in a harsh environment. Therefore, the wording has been revised to state the trip setpoints consider "environmental effects."

P87 NUREG Table 3.3.6.1-1 Functions 2.d (Reactor Building Exhaust Radiation-High) and 2.e (Refueling Floor Exhaust Radiation-High) have not been included as a primary containment isolation function. Although these functions will initiate isolation of certain pipelines, no credit is taken for them to perform this function. The primary containment high pressure and reactor vessel low water level functions are adequate in effecting appropriate isolation of the same pipelines. The Reactor Building ventilation exhaust high radiation isolation is provide only as a third, redundant method of detecting breaks in nuclear system process barrier significant enough to require automatic isolation. This function has been included as required Secondary Containment Isolation Instrumentation and Control Room Emergency Ventilation (CREV) Initiation Instrumentation.

P88 BFN has not previously implemented the technical specification improvements recommended in Reference 9 and reflected in the surveillance intervals for RPS Instrumentation documented in NUREG-1433. Addition of RPS Channel Test Switches to the list of functions assures weekly testing of the RPS scram contactors. Weekly testing of the scram contactors was credited in the analysis of Reference 9 to extend many automatic scram functions' frequencies.

P89 BFN's Average Power Range Monitor design includes only two flow units, one for the APRMs assigned to RPS A and the other assigned to RPS B. A flow unit cannot be taken out of service for calibration without causing a trip of multiple APRMs. Because of this design, the flow units are calibrated only during refueling outages. Therefore, the frequency of the NUREG SR has been changed to once per 18 months.

P90 BFN plans to implement the Power Range Neutron Monitor (PRNM) upgrade including the ARTS (APRM/RBM Technical Specification) Improvement Program for Unit 2 in the fall of 1997 and for Unit 3 in the fall of 1998. In support of those modifications Technical Specification (TS) Change Request 353 was submitted by letter dated June 2, 1995. The original TS Change Request 353 was applicable to the CTS only. The submittal was based in part on NEDC-32433P "Maximum Extended Load Line Limit and Arts Improvement Program Analyses for Browns Ferry Nuclear Plant Unit 1, 2, and 3. On April 11, 1997, TS Change Request 353S1 was submitted for the necessary changes to the ITS necessary due to the

modifications for PRNM upgrade. By letter dated September 11, 1997 Amendment No. 249, including a Safety Evaluation, was received approving the CTS changes for PRNM for BFN Unit 2. The MCPR values for RBM in footnote (a) and (b) to Table 3.3.2.1 are the same as those approved in the SER. BFN intends to resubmit TS Change Request 353S1 with any changes necessary resulting from NRC questions on the ITS. Since the original ITS submittal was based on not having the PRNM implemented the setpoints will continue to be flow biased until PRNM incorporation. BFN chose to utilize the power dependent setpoint values provided in NEDC-32433P as flow biased values in the ITS based on the Rod Withdrawal Error (RWE) analysis included in NEDC-32433P. The RWE analysis states "the three points analyzed are considered sufficient to bound all expected power/flow states." The use of the power based setpoints values as flow based setpoints are more restrictive and are addressed in JFC M3. Those functions not yet implemented by PRNM Intermediate Power Range - Upscale, High Power Range - Upscale, and Bypass Time Delay were deleted from the ITS. The remaining changes to the ITS necessary due to the ARTS upgrade are addressed in TS Change Requests 353 and 353S1. Based on the preceding information it is concluded that it is acceptable to utilize the power based setpoint values as flow biased setpoints and defer the other necessary changes until implementation of PRNM.

P91 Proposed SR 3.3.1.1.9 has been added to address plant specific calibration requirements for RPS instrumentation. A BFN specific setpoint analysis determined that a calibration frequency of three months (92 days) is required for the IRM and APRM functions (1.a, 2.a, 2.b, and 2.c) that NUREG-1433 specifies SR 3.3.1.1.13 (required calibration frequency of 18 months) and SR 3.3.1.1.11 (required calibration frequency of 184 days). Based on this, SR 3.3.1.1.9 was necessary to provide a calibration frequency of 92 days for those functions. Appropriate Bases changes have also been made.

P92 Not used.

P93 Deleted (Per NRC Open Item 3.3.6.1-21).

P94 Deleted (Per NRC Open Item 3.3.6.1-22).

P95 NUREG-1433, Table 3.3.6.1-1, Note (c) specifies allowance for the condition of maintaining Shutdown Cooling System Isolation capability without single failure protection when RHR Shutdown Cooling System integrity is maintained. The Note (proposed ISTS Table 3.3.6.1-1,Note (b)) has been revised to specify this condition based on BFN design. The BFN design is such that isolation capability is maintained when one channel per trip system for one RHR Shutdown Cooling (SDC) Supply isolation valve is OPERABLE. Under these conditions (Modes 4 and 5 with RHR Shutdown Cooling System integrity is maintained), the LPCI to

Reactor isolation valves are not required to receive isolation signals from this Function since the check valves (PCIVs) in these penetration flow paths provide for isolation without single failure protection.

P96 Proposed ISTS 3.3.6.1 Function 6.c, Drywell Pressure - High Function, has been added since the BFN design is such that this Function also provides for Shutdown Cooling System Isolation. Appropriate changes to the NUREG-1433 specification and Bases have been made to address this change.

P97 A statement that the LOGIC SYSTEM FUNCTIONAL TEST shall include a calibration of time delay relays and timers necessary for proper functioning of the logic has been added to the Bases for proposed SR 3.3.6.1.6. The addition of this statement is consistent with the CTS (NOTES FOR TABLES 4.2.A THROUGH 4.2.L EXCEPT 4.2.D AND 4.2.K, Note 6).

P98 A statement has been added to the Bases for proposed SR 3.3.6.2.2 and SR 3.3.7.1.2 to indicate that the Functional Test for proposed ISTS 3.3.6.2 and 3.3.7.1 Functions 3 and 4 shall consist of verifying the High Voltage Power Supply (HVPS) voltage at the Sensor and Convertors (detectors) is within its design limits, and that a CHANNEL FUNCTIONAL TEST as defined in Section 1.1, "Definitions" shall be performed once per 18 months as part of the CHANNEL CALIBRATION for proposed ISTS 3.3.6.2 and 3.3.7.1 Functions 3 and 4. The addition of this statement is consistent with the CTS (NOTES FOR TABLES 4.2.A THROUGH 4.2.L EXCEPT 4.2.D AND 4.2.K, Note 30 for Units 1 and 3 and Note 32 for Unit 2).

P99 For BFN the ECCS instrumentation (i.e., the CS logic) supports generation of the Common Accident Signals (CAS) which initiates the start of all DG's and EECW System. For conditions of Reactor Vessel Water Level - Low Low Low (Level 1), or Drywell Pressure - High with Reactor Steam Dome Pressure - Low (Injection Permissives and ECCS Initiation) each CS trip system initiates start of its associated CS Subsystem and generates a CAS signal (i.e., one CS trip system generates a CAS A signal and the other will initiate a CAS B signal. The CAS signal generated from either CS trip signal will initiate the start of all DGs and EECW pumps. Considering the BFN design for initiation of the DGs and EECW pumps, the inclusion of these features within LCO 3.3.5.1 results in over restrictive actions. For example, if one channel or Reactor Steam Dome Pressure - Low (Injection Permissive and ECCS Initiation) was inoperable for more than 24 hours the Completion time for Required Action C.2 would not be met and Condition H.1 would be entered. Required Action H.1 would appropriately require the support ECCS pumps to be declared inoperable (an additional single failure would prevent initiation of the associated ECCS pumps). However, if the DGs and EECW pumps were also included as support features, the declaration of these features inoperable would not be appropriate since an

additional single failure could not prevent their initiation. Additionally, during operation the most limiting Required Action which would be entered as a result of declaring all associated supported features inoperable would be the Required Action of LCO 3.5.1. Since each of the instruments for the three variables are associated with at least one CS trip system and both LPCI trip systems, declaration of supported features inoperable would result in Required Action H.1 of LCO 3.5.1 being entered which requires LCO 3.0.3 be entered immediately. Declaration of all DGs inoperable would result in Required Action G.1 of LCO 3.8.1 being entered which would allow two hours to restore the DGs to OPERABLE status before Condition H was entered (LCO 3.8.1 Required Actions G.1, H.1, and H.2 result in no specified time to be entered to be in MODE 2 and extend the completion time to be in MODE 3 and 4 by one hour as compared to LCO 3.0.3). Declaration of all EECW pumps inoperable would result in Condition B of LCO 3.7.2 being entered. The Required Actions for Condition B of Lco 3.7.2 require the plant to be in MODE 2 but decrease the overall completion time to be in MODE 3 and 4 by one hour as compared to LCO 3.0.3). The Common Accident Signal logic is addressed in proposed LCO 3.8.1. When one division of the CAS log is inoperable, Required Action D.1 of LCO 3.8.1 requires the division of logic to be returned to OPERABLE status in seven days and when both divisions of CAS logic are inoperable, Required Action I.1 requires LCO 3.0.3 to be entered immediately. Based on the above, BFN proposes to limit the actions of LCO 3.3.5.1 to supported ECCS features only.

P100 BFN design is such that there is no logic system (no logic relays, etc. other than the flow switch) for actuation of the Core Spray pump minimum flow valves. Based on this design, the SR for performance of a LOGIC SYSTEM FUNCTIONAL TEST has not been included for Function 1.d (CS Pump Discharge Flow Low (Bypass)).

BROWNS FERRY NUCLEAR PLANT - IMPROVED TECHNICAL SPECIFICATIONS
SECTION 3.3
REVISION 2
LIST OF REVISED PAGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS

Replaced pages 1 of 68 through 68 of 68 Revision 1 with pages 1 of 72 through 72 of 72 Revision 2

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will remove normal operation requirements for operability of RPS functions in MODE 5 unless a control rod is withdrawn from a core cell containing fuel assemblies. Control rod withdrawal from core cells without fuel assemblies does not significantly affect core reactivity and the RPS scram function serves no useful purpose. The affected RPS functions are not considered as initiators for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, this change does not impact the capability of the system to perform its required function, i.e. insert withdrawn control rods. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the requirements continue to provide operability of the RPS scram functions under any conditions where they may be required.

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will remove normal operation requirements for operability of RPS functions in MODEs 3 and 4. Control rod withdrawal is not allowed in these conditions and the RPS scram function serves no useful purpose. These RPS functions are not considered as initiators for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, this change does not impact the capability of the system to perform its required function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since other requirements prevent the removal of control rods in these modes which renders the RPS scram functions moot.

"L3" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware changes. The RPS Instrumentation is not assumed to be an initiator of any analyzed event. The change will not allow continuous operation such that a single failure will preclude the affected channel's function from being performed. This change allows an additional 8 hours to reach MODE 3, which provides a reasonable amount of time to perform an orderly shutdown, thus further minimizing a potential upset from a too rapid decrease in plant power. Additionally, the consequences of an event occurring while the unit is being shutdown during the extra 8 hours is the same as the consequences of an event occurring for the current 4 hours. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The increased time allowed for reaching MODE 3 with inoperable RPS channels is acceptable based on the small probability of an event requiring the inoperable channels to function and the desire to minimize plant transients. The requested 8 hour extension will provide sufficient time for the unit to reach MODE 3 in an orderly manner. As a result, the potential for human error will be reduced. As such, any reduction in a margin of safety will be insignificant and offset by the benefit gained from providing sufficient time to reach MODE 3, thus avoiding potential plant transients from attempting to reach MODE 3 in the current time.

"L4" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve a hardware change. The APRM RPS instrumentation is not assumed in the initiation of any analyzed event. The role of this instrumentation is in mitigating and, thereby, limiting the consequences of analyzed events. The proposed change effectively extends the initial Surveillance Frequency until 12 hours after Thermal Power is ≥ 25% RTP. This allows time after the appropriate conditions are established to perform the Surveillance. The Surveillance is not required to be performed below 25% RTP because it is difficult to accurately determine core Thermal Power from a heat balance at these low power levels. In addition, at low power levels, a high degree of accuracy between the APRM indication and actual core Thermal Power is unnecessary due to the large inherent margin to the thermal limits at these power levels. As a result, the consequences of an accident are not affected by this change. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This change will not alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated

"L4" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The margin of safety is not reduced by this change since the proposed change to the Surveillance Frequency provides the necessary assurance that the APRM instrumentation has been accurately calibrated at the earliest opportunity. This change extends the initial performance of the Surveillance Requirement to within 12 hours after reaching 25% RTP. This is considered acceptable since below 25% RTP a high degree of accuracy between the APRM indication and actual core Thermal Power is unnecessary due to the large inherent margin to the thermal limits at these power levels. In addition, this change provides the benefit of allowing the Surveillance to be postponed until appropriate plant conditions exist for performing the Surveillance accurately. The safety analysis assumptions will still be maintained, thus no question of safety exists. Therefore, this change does not involve a significant reduction in a margin safety.

"L5" CHANGE

TVA has concluded that operation of BFN in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change (Proposed BFN ISTS 3.3.1.1, Required Action C.1) provides a 1 hour Completion Time from discovery of loss of trip capability prior to requiring further action. Further action is not required by C.1 unless trip capability is lost. This is acceptable since the Function is still capable of performing its design function. The current wording of Note 1 to CTS Table 3.1.A could require further action be initiated immediately even if trip capability is maintained. The change does not result in any hardware changes. RPS instrumentation are not assumed to be initiators of any analyzed event. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in trip capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   Currently, no specific time period is allowed to initiate further required action upon discovering a loss of trip capability. The proposed change allows 1 hour, which is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping channels. Proposed required action C.1 does not require further action unless trip capability is lost. This is acceptable since the Function is still capable of performing its design function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

"L6" CHANGE

TVA has concluded that operation of BFN in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware changes. The requested change is to delete CTS Action 1.A for Table 3.1.A which requires control rods to be inserted in four hours for the APRM flow bias, APRM fixed trip, APRM downscale,MSIV closure, Turbine Control Valve Fast Closure or Turbine Trip, and Turbine Stop Valve Closure trip functions. For each of the affected functions Action 1.A is or'd with another Action (1.B, 1.C, or 1.D) which is the preferred option. These preferred options are maintained in the ITS and thus the actions to be taken would be the same since a reduction in power is preferred over shutting down. Thus, since option (Action) 1.A would in most cases not have been used the change in actual plant operation due to its deletion is not appreciable. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not 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 change in parameters governing normal plant operation. The proposed change will delete CTS Action 1.A for Table 3.1.A which requires control rods to be inserted in four hours for the APRM flow bias, APRM fixed trip, APRM downscale, MSIV closure, Turbine Control Valve Fast Closure or Turbine Trip, and Turbine Stop Valve Closure trip functions in favor of keeping less restrictive Actions which are also currently options. Since no new or different accident initiators are created, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The proposed change in the deletion of CTS Action 1.A for Table 3.1.A which requires control rods to be inserted in four hours for the APRM flow bias, APRM fixed trip, APRM downscale, MSIV closure, Turbine Control Valve Fast Closure or Turbine Trip, and Turbine Stop Valve

"L6" CHANGE (continued)

Closure trip functions does not produce a significant reduction in a margin of safety because the alternate Actions place the plant in a condition which adequately compensates for the loss of the affected functions.

"L7" CHANGE

TVA has concluded that operation of BFN in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change is to delete the requirement to verify IRM/APRM overlap during a controlled startup and replace it with a requirement to verify IRM/APRM overlap when entering MODE 2 from MODE 1 (during shutdown). This change does not involve a significant increase in the probability or consequences of an accident previously evaluated because during startup system design will prevent reactivity increases by initiating a rod block if inadequate overlap is not maintained.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The change does not involve a significant reduction in a margin of safety since the system design will not permit operation if adequate overlap does not exist during startup and IRM functionality will now be verified prior to entering a condition where it is necessary to monitor reactivity during shutdown.

"L8" CHANGE

TVA has concluded that operation of BFN in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change is to delete the requirement for the APRM High Flux (Setdown) - (≤ 15% rated power) function and APRM High Flux Inoperative function to be Operable in the Refuel Mode. These APRM functions cannot initiate an analyzed accident. Therefore, the probability of an accident previously analyzed is not increased. The IRMs are capable of performing the required trip function and the SRMs are capable of indicating impending criticality and producing a Rod Block Signal. Therefore the consequences of an accident previously analyzed are not significantly increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not create a new mode of plant operation nor does it involve any plant modifications. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The IRMs perform the same trip function as the APRM functions and additional indication and rod block capability circuitry is available from the SRMs. In addition, only one rod withdrawal is permitted in the Refuel Mode and there is adequate margin to assure the core remains noncritical with one rod out and withdrawal of a second rod is prevented by refueling interlocks and administrative controls. Therefore, the change does not involve a significant reduction in a margin of safety.

"L9" CHANGE

TVA has concluded that operation of BFN in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not involve any safety system and does not affect the operation of any equipment. The change does not affect the fundamental method by which the LPRMs are calibrated. The increased time between required LPRM calibrations does not affect either the initiator of any accident previously evaluated or any equipment required to mitigate the consequences of an accident, or the isotopic inventory in the fuel. Thus, the change does not increase either the probability or the radiological consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not introduce a new mode of plant operation and does not involve the installation of any new equipment modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The proposed change to lengthen the period between LPRM calibrations could affect the accuracy of the thermal power distribution as determined by the LPRM System. Outputs from the LPRMs are used in the Reactor Protection System (Average Power Range Monitors), Rod Block Monitor, Rod Worth Minimizer, and for daily monitoring for compliance with power distribution limits, (i.e., plant thermal margins).

   Accuracy requirements on the power distribution are defined by GESTAR-II and "GE Fuel Bundle Designs" which are part of the licensing basis. In particular, Table 3-3 of "GE Fuel Bundle Designs" (NEDE-3112P) requires TIP readings to have a root mean square (rms) nodal power uncertainty of no more than 8.7% for reload cores. The attending Table 3-3 comment states that this uncertainty also applies to the power distribution as determined by the LPRM System. Thus, the accuracy in power distribution as determined by the LPRM System between TIP sets must also meet the 8.7% rms uncertainty.

"L9" CHANGE (continued)

This 8.7% nodal uncertainty reflects a statistical combination of several components, two of which are the uncertainty associated with the nuclear model methodologies and the uncertainty associated with the LPRM calibration interval. Advances in process computer monitoring include the development of new mathematical techniques and algorithms combining reactor physics theory with online core data, (e.g., LPRM readings). One such methodology presently employs an adaptive learning algorithm using online and historic core data inputs to improve power calculations within the reactor physics model by effectively modifying the neutron leakage terms to force the calculated power distribution to match the measured power distribution as determined by the TIP System.

Subsequent calculations of power distribution are modified by the adaptive coefficients and LPRM readings during monitoring between LPRM calibrations.

Thus, there is no reduction in accuracy of the power distribution inputs to the Reactor Protection System, the Rod Block Monitor, the Rod Worth Minimizer or compliance monitoring with daily power distribution limits. Thus, there is no reduction in the margin of safety associated with any of these systems occasioned by this change.

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   If a spiral offload or reload pattern Is used, the proposed specifications will allow a reduction in the number of SRM channels required to be operable during refueling. The probability of an accident is not increased by relaxed SRM operability requirements when using a spiral pattern for fuel movements because the use of a spiral pattern provides assurance that the SRM will be in the optimum position for monitoring changes in neutron flux levels resulting from the Core Alteration. The consequences of an accident will not be increased by these changes because the SRMs are not credited for the mitigation of any accidents. The APRM Flux scram and not any SRM function is credited for mitigating a rod withdrawal or reactivity addition accident. Backup to the APRM Flux scram during excessive reactivity additions is provided by IRM rod blocks and IRM Range 1 High Flux Trip. Additionally, the reactivity addition accidents are assumed to be initiated from below the level of source range detector sensitivity and, therefore, are independent of any changes in the ability to monitor changes In the source range flux level. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L1" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

If a spiral offload or reload pattern is used, the proposed specifications will allow a reduction in the number of SRM channels required to be operable during refueling. The proposed change does not involve a significant reduction in a margin of safety because: SRMs are not credited in any safety analysis; at least one SRM will remain Operable during rod withdrawal; and, the use of a spiral pattern provides assurance that the SRM will be in the optimum position for monitoring changes in neutron flux levels resulting from the Core Alteration. As a result, the change does not affect the current analysis assumptions. Therefore, this change does not Involve a significant reduction in a margin of safety.

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change is to adopt Note (a) to ITS Table 3.3.1.2-1 which modifies the applicability for the SRMs in Mode 2 to when the IRMs are on Range 2 and below. This is acceptable because at the flux levels for which the IRMs are on Range 3 and above, the IRMs are the appropriate Neutron Monitors. The SRMs are not assumed to operate during any design basis accident or transient analyzed in the FSAR and therefore their applicability will not involve a significant increase in the probability or consequences of an accident previously analyzed.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not create a new mode of plant operation nor does it involve any plant modifications. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The IRMs perform the same indication function as the SRMs during the time overlap occurs. The withdrawal of the SRMs was not specifically called for in the CTS but is consistent with the FSAR. Therefore, the change does not involve a significant reduction in a margin of safety.

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Proposed Surveillances for the Rod Block Monitor will allow entry in the Conditions and Required Actions for inoperable RBM to be delayed up to six hours provided the redundant channel maintains rod block capability. The purpose of the RBM is to limit a rod withdrawal error (RWE) and prevent violation of the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) and the fuel cladding design limit of less than 1% plastic strain. The probability of an accident is not increased because the Note applies only when the rod block function is maintained by the redundant RBM channel, the six hour period is shorter than the allowable out of service time (AOT) of 24 hours, and tests are conducted infrequently. The consequences of an accident will not be increased because the purpose of the RBM is to limit the a rod withdrawal error (RWE) and prevent violation of the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) and the fuel cladding design limit of less than 1% plastic strain and has no function in mitigating the consequences of an accident. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L1" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Proposed Surveillances for the Rod Block Monitor will allow entry in the Conditions and Required Actions for inoperable RBM to be delayed up to six hours provided the redundant RBM channel maintains rod block capability. This change does not involve a significant reduction in a margin of safety because this Note applies only when the rod block function is maintained by the redundant RBM channel, the six hour period is shorter than the allowable out of service time (AOT) of 24 hours, and tests are conducted infrequently. Additionally, 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, modified, or tested. As a result, the change does not affect the current analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change is less restrictive in two ways. First, the existing specifications require performance of the Channel Functional Tests "prior to" reaching the condition where the RWM is required to be Operable (except for CTS 4.3.B.3.b.2.c) but the proposed Surveillance Tests are "not required until one hour after" reaching the condition where the RWM is required to be Operable. Second, the existing specification requires that the Channel Functional Test be performed on every startup and shutdown but the proposed specifications will require Channel Functional Tests on startups and shutdowns only if the test has not been performed In the previous 92 days. The purpose of the RWM is to limit a rod withdrawal error (RWE) and prevent violation of the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) and the fuel cladding design limit of less than 1% plastic strain. The change does not allow continued operation in a configuration such that a single failure will result in the loss of the control rod block function. In addition, the probability of an accident is not increased because: a) the Rod Worth Minimizer does not monitor core thermal conditions but simply enforces preprogrammed rod patterns as a backup intended to prevent reactor operator error in selecting or positioning control rods; b) reliability analysis documented in NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988 determined that the failure frequency curve for this instrumentation is relatively flat in the range of 30 to 124 days and starts a gradual increase after 124 days which means that more frequent testing is unlikely to identify problems; and, c) it is overly conservative to assume that the RWM is not operable when a surveillance is not performed because of its demonstrated reliability as demonstrated by successful completion of most Channel Functional Tests. The consequences of an accident will not be increased because the RWM is intended to prevent exceeding thermal limits and has no function in mitigating the consequences of an accident. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

"L2" CHANGE (continued)

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The proposed change does not involve a significant reduction in a margin of safety because: a) the Rod Worth Minimizer does not monitor core thermal conditions but simply enforces preprogrammed rod patterns as a backup intended to prevent reactor operator error in selecting or positioning control rods; b) reliability analysis determined that the failure frequency curve for this instrumentation is relatively flat in the range of 30 to 124 days and starts a gradual increase after 124 days which means that frequent testing is unlikely to identify problems; and, c) it is overly conservative to assume that the RWM is not operable when a surveillance is not performed because of its demonstrated reliability as demonstrated by successful completion of most Channel Functional Tests. Therefore, this change does not involve a significant reduction in a margin of safety.

"L3" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change will extend the Completion Time for blocking control rod withdrawal if one RBM channel is inoperable from immediately to within 25 hours. Additionally, proposed LCO 3.3.2.1, Condition B, will extend the Completion Time for blocking control rod withdrawal if both RBM channels are inoperable from immediately to within 1 hours. However, the requirement to block control rod withdrawal if a RBM channel is inoperable will exist whenever the RBM function is required to be OPERABLE and not just "during operation with CMFCP or CMFLPD equal to or greater than 0.95" and prevent violation of the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) and the fuel cladding design limit of less than 1% plastic strain. During the 24 hours of operation permitted with one RBM channel inoperable, the remaining Operable channel is adequate to perform the control rod block function. During the 1 hour of operation permitted with both RBM channels inoperable and a complete loss of the RBM function, a rod withdrawal error is unlikely while allowing time for restoration or tripping of inoperable channels. In both cases, continued operation in a configuration such that a single failure will result in the loss of the control rod block function is strictly limited. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L3" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change will extend the Completion Time for blocking control rod withdrawal if one RBM channel is inoperable from immediately to within 25 hours. Additionally, proposed LCO 3.3.2.1, Condition B, will extend the Completion Time for blocking control rod withdrawal if both RBM channels are inoperable from immediately to within 1 hour. However, the requirement to block control rod withdrawal if a RBM channel is inoperable will exist whenever the RBM function is required to be Operable and not just "during operation with CMFCP or CMFLPD equal to or greater than 0.95" as is required by existing Specification 3.3.B.5. This change does not involve a significant reduction in a margin of safety because during the 24 hours of operation permitted with one RBM channel inoperable, the remaining Operable channel is adequate to perform the control rod block function. During the 1 hour of operation permitted with both RBM channels inoperable and a complete loss of the RBM function, a rod withdrawal error is unlikely while allowing time for restoration or tripping of inoperable channels. In both cases, continued operation in a configuration such that a single failure will result in the loss of the control rod block function is strictly limited. As a result, the change does not affect the current analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

"L4" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change eliminates Specification 4.3.B.5 which requires a Functional Test of the Rod Block Monitor (RBM) "prior to withdrawal of the designated rod(s)" during operation with CMFCP or CMFLPD equal to or greater than 0.95 and relies completely upon the Functional Test which is required every 92 days. The probability of an accident is not increased by this change because: two independent RBM channels will be Operable during any rod withdrawal except for short and infrequent periods when one channel is inoperable; and, deletion of this requirement allows taking credit for routine periodic tests in place of performing unscheduled testing whenever the potential exists that the RBM may be required to function. The Frequency of 92 days for the Channel Functional Test is based upon the reliability analysis in NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation, "October 1988. This reliability study found that the failure frequency curve for this type of instrumentation is relatively flat in the range of 30 to 124 days and starts a gradual increase after 124 days. Based on this finding, performing this testing more frequently than every 92 days does not significantly increase the probability of detecting a random failure of the RBM. The consequences of an accident will not be increased because the purpose of the RBM is to limit the a rod withdrawal error (RWE) and prevent violation of the Minimum Critical Power Ratio (MCPR) Safety Limit (SL) and the fuel cladding design limit of less than 1% plastic strain and has no function in mitigating the consequences of an accident, Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L4" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change eliminates Specification 4.3.B.5 which requires a Functional Test of the Rod Block Monitor (RBM) "prior to withdrawal of the designated rod(s)" during operation with CMFCP or CMFLPD equal to or greater than 0.95 and relies completely upon the Functional Test which is required every 92 days. The proposed change does not involve a significant reduction in a margin of safety because: two independent RBM channels will be OPERABLE during any rod withdrawal except for short and infrequent periods when one channel is inoperable; and, deletion of this requirement allows taking credit for routine periodic tests in place of performing unscheduled testing whenever the potential exists that the RBM may be required to function. The Frequency of 92 days for the Channel Functional Test is based upon the reliability analysis in NEDC-30851-P-A, "Technical Specification Improvement Analysis for BWR Control Rod Block Instrumentation," October 1988. This reliability study found that the failure frequency curve for this type of instrumentation is relatively flat in the range of 30 to 124 days and starts a gradual increase after 124 days. Based on this finding, performing this testing more frequently than every 92 days does not significantly increase the probability of detecting a random failure of the RBM. As a result, the change does not affect the current analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.3.1
POST-ACCIDENT MONITORING INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The change will increase the surveillance interval to allow conduct of a CHANNEL CHECK once every 31 days and a CHANNEL CALIBRATION once every 18 months. The affected instruments are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, an increase of the surveillance interval will not affect the capability of the component or system to perform its functions (i.e., its readability). Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since experience has shown that similar components usually pass the Surveillance and maintain necessary accuracy when performed at the proposed frequency.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.3.1
POST-ACCIDENT MONITORING INSTRUMENTATION

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will revise the Required Actions for inoperable PAMs that are not restored to service within the allowed out-of-service time. PAMs are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Also, this change does not further degrade the capability of the system to perform its required function under these circumstances. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the monitors are not required to provide automatic response to any design basis accident. The additional time has been evaluated and determined to not significantly affect the contribution of the monitors to risk reduction.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.3.1
POST-ACCIDENT MONITORING INSTRUMENTATION

"L3" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will limit the required applicability for post accident monitors to those MODES during which design basis events are assumed to occur. PAMs are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Also, this change does not degrade the capability of the system to perform its design basis accident function. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the monitors are provided to assist the response to design basis accidents in the MODES which continue to be applicable. The low probability of such events in the other MODES assures that any impact on the margin of safety is insignificant.

"L4" Change

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware changes. The hydrogen analyzer channels are not assumed to be initiators of any analyzed event. The role of this instrumentation is in providing the operators information after an accident to allow them to take mitigating actions, thereby limiting consequences. The requested change does not allow continuous operation since the available alternate indications may not fully met all performance qualification requirements applied to PAM instrumentation. The change allows additional time to restore the inoperable analyzers, thus minimizing the potential for a shutdown transient. Additionally, the consequences of an event occurring during the proposed allowed outage time are the same as the consequences of an event occurring during the existing allowed outage times. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not 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 change in parameters governing normal plant operation. The proposed change will allow 72 hours to restore a single hydrogen analyzer when two are inoperable. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The proposed allowed outage time for hydrogen analyzers are acceptable based on the small probability of an event requiring the hydrogen analyzers during the 72 hour time period, the passive function of the analyzers and the availability of alternate means to obtain the information. Providing a 72 hour allowed outage time for two inoperable hydrogen analyzers will minimize the potential for plant transients that can occur during shutdown by providing additional time to restore the

"L4" Change (continued)

analyzers. As such, any reduction in a margin of safety by the extension of the allowed outage time will be offset by the benefit gained by avoiding an unnecessary plant shutdown transient. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.3.1
POST-ACCIDENT MONITORING INSTRUMENTATION

"L5" Change

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not result in any hardware changes. The Primary Containment Area Radiation channels are not assumed to be initiators of any analyzed event. The role of this instrumentation is in providing the operators information relative to primary containment radiation levels during and after an accident to allow them to take mitigating actions, thereby limiting consequences. Additionally, the consequences of an event occurring during the proposed 30 day allowed outage time are the same as the consequences of an event occurring during the current allowed outage time. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not 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 allow 30 days for restoration of the inoperable channel or initiation of an alternate means of monitoring primary containment radiation. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The proposed 30 day Completion Time to restore a Primary Containment Area Radiation channel to Operable status is acceptable based on the passive nature of the instruments, the remaining Operable or diverse instrument channels and the small probability of an event requiring the Primary Containment Area Radiation channel during this time period. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.3.1
POST-ACCIDENT MONITORING INSTRUMENTATION

"L6" Change

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware changes. The Primary Containment Area initiators of any analyzed event. The role of this instrumentation is in providing the operators information relative to primary containment radiation levels during and after an accident to allow them to take mitigating actions, thereby limiting consequences. The requested change does not allow continuous operation since the available alternate indications may not fully meet all performance qualification requirements applied to the Primary Containment Area Radiation channels. The change allows 7 days to restore one inoperable channel or to initiate the alternate method of monitoring, thus minimizing the potential for a shutdown transient. Additionally, the consequences of an event occurring with the proposed actions are the same as the consequences of an event occurring within the allowed outage time of the current actions. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not 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, when two monitor channels are inoperable, will allow 7 days to restore one inoperable channel or initiate an alternate means of monitoring primary containment radiation. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The proposed change to allow 7 days to restore one Primary Containment Area Radiation channel to Operable status or initiate alternate means of monitoring is acceptable based on the small probability of an event requiring the Primary Containment Area Radiation channels during the

"L6" Change (continued)

time period, the passive nature of the monitors, and the availability of alternate means to obtain the required information. Providing the proposed action will minimize the potential for plant transients that can occur during shutdown by providing additional time for the restoration of one monitor or the initiation of an alternate means of monitoring. As such, any reduction in a margin of safety resulting from the proposed change will be offset by the benefit gained by avoiding an unnecessary plant shutdown transient. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.3.1
POST-ACCIDENT MONITORING INSTRUMENTATION

"L7" Change

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change relaxes the time allowed to submit a report after the allowed outage time for an inoperable Primary Containment Area Radiation Monitor channel(s) expires from within 7 days to within 14 days. This is consistent with the requirements in NUREG-1433. This change will not result in operation that will increase the probability of initiating an analyzed event since the time frame for submitting a Special Report is not assumed in the initiation of any analyzed event. This change only affects the time frame for submitting the report after the allowed outage time for an inoperable channel(s) has expired. This change will not alter assumptions relative to mitigation of an accident or transient event. This change will not alter the operation of process variables, structures, systems, or components as described In the safety analyses. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   This change relaxes the requirement for submitting a Special Report to the NRC. This change will not alter the plant configuration (no new or different type of equipment will be installed). This change only affects the time allowed to submit a report. This change does not impose different requirements; a report is still required. It will not alter assumptions made in the safety analysis. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L7" Change (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

This change proposes to relax the time required for submittal of the report. The time is extended from 7 days to 14 days after the allowed outage time for an inoperable channel(s) expires. Increasing the time for submitting a report does not affect the margin of safety since this change will not impact any safety analysis assumptions. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

"L8" Change

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware changes. The hydrogen analyzer channels are not assumed to be initiators of any analyzed event. The role of the affected instrumentation is to provide the operators information after an accident to allow the to take mitigating actions, thereby limiting the consequences. The requested change is to the frequency of the CHANNEL FUNCTIONAL TESTS which are to be incorporated into the CHANNEL CALIBRATIONS and supplemented by the addition of CHANNEL CHECKS. The existing CHANNEL FUNCTIONAL TESTS are performed monthly. The Proposed CHANNEL CALIBRATIONS will be performed quarterly (92 days) and the Proposed CHANNEL CHECKS will be performed monthly (31 days). Since the CHANNEL CHECKS will be performed at the same frequency as the current CHANNEL FUNCTIONAL TESTS, and both confirm operability of the channels, the time a channel may be inoperable without being identified will not increase significantly. In addition, the CHANNEL FUNCTIONAL TESTS will still be performed as part of the CHANNEL CALIBRATIONS. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not 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 change in parameters governing normal plant operation. The proposed change will allow relaxation of the frequency of CHANNEL FUNCTIONAL TESTS from monthly to quarterly which is compensated for by adding a monthly CHANNEL CHECK. Since no new or different accident initiators are created, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L8" Change (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed change in the frequency of CHANNEL FUNCTIONAL TESTS is acceptable based on hydrogen analyzer channels overall operability being maintained at the essentially the same level by the addition of other means of verifying operability. The function of providing indication for these channels can be confirmed by observation as any malfunction will be apparent by differences between the indicated values. As such the reduction in a margin of safety by the change in frequency of performing CHANNEL FUNCTIONAL TESTS will be compensated for by performing CHANNEL CHECKS and CHANNEL CALIBRATIONS. Therefore, this change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.4.1 - EOC-RPT SYSTEM INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will entry into associated Conditions and Required Actions to be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability when a channel is placed in inoperable status solely for performance of required Surveillances. Since trip capability is maintained the instrumentation continues to provide the required safety function. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the safety functions continue to provide the required ATWS-RPT actuation capability.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.4.1 - EOC-RPT SYSTEM INSTRUMENTATION

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow an inoperable channel to be placed in the tripped condition to satisfy the Required Actions and allow continued operation. A tripped channel continues to provide the required safety function. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the safety functions continue to provide the required ATWS-RPT actuation capability, including single failure conditions.

"L3" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow two hours to restore EOC-RPT trip capability when one or more EOC-RPT trip functions are not maintained. The operation of these trip functions is not a precursor to any design basis accident or transient analyzed in the Browns Ferry Updated Final Safety Analysis Report. Therefore, this change does not increase the probability of any previously evaluated accident. The 2 hour Completion Time is sufficient time for the operator to take corrective action, and takes into account the likelihood of an event requiring actuation of the EOC-RPT instrumentation during this period. It is also consistent with the 2 hour Completion Time provided in LCO 3.2.2, Minimum Critical Power Ratio (MCPR) for Required Action A.1, since this instrumentation's purpose is to preclude a MCPR violation. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The EOC-RPT trip functions are designed to trip the recirculation pumps in the event of a turbine trip or generator load rejection to mitigate the increase in neutron flux, heat flux, and reactor pressure, and to increase the margin of the MCPR safety limit. This change does not involve a significant reduction in a margin of safety since the trip functions continue to provide the required EOC-RPT actuation capability, including single failure conditions that is consistent with proposed LCO 3.2.2, MCPR, and current Technical Specification 3.5.K, which allow two hours to restore MCPR limits when not within limits.

"L4" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change provides an LCO requirement to allow the MCPR limit of LCO 3.2.2 to be reduced by an appropriate amount (specified in the COLR) if the instrumentation is inoperable. This change does not result in any hardware or operating procedure changes. The EOC-RPT instrumentation is not assumed to be an initiator of any analyzed event. The instrumentation's role is in mitigating and thereby limiting the consequences of violating the MCPR Safety Limit during a turbine trip or generator load rejection. Therefore, providing a MCPR operating limit reduction provides the same level of protection as the instrumentation. Thus, operation with EOC-RPT inoperable but the MCPR limits for EOC-RPT inoperable met, will have the same consequences in the event of a design basis transient as with the EOC-RPT OPERABLE. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated is involved.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not involve any design changes, plant modifications, or changes in plant operation. The overall system will continue to function in the same way as before the change (i.e., MCPR Safety Limit will be protected). Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The MCPR operating limit reduction provides the same level of protection as the current EOC-RPT System Instrumentation. Therefore, the change does not involve any reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.4.2 - ATWS-RPT INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Proposed Required Actions A.1 and A.2 allow 14 days to declare the trip system inoperable or place the channel in a tripped condition immediately. Because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse Functions, and the low probability of an event requiring the initiation of ATWS-RPT, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The ATWS-RPT trip functions are designed to trip the recirculation pumps following events where a scram does not (but should) occur, to lessen the effects of an ATWS event. With one or more channels inoperable, but with the ATWS-RPT capability for each Function maintained, the ATWS-RPT System is capable of performing its intended function. Fourteen days is allowed to restore the inoperable channel or place it in the tripped condition because of the diversity of sensors available to provide trip signals, the low probability of extensive numbers of inoperabilities affecting all diverse functions, and the low probability of an event requiring the ATWS-RPT. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.4.2 - ATWS-RPT INSTRUMENTATION

"L2" Change

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow one hour to restore ATWS-RPT trip capability for one Function prior to initiating entry into MODE 2. The one hour period allows sufficient time for the operator to take corrective actions. The probability of an event occurring during this one hour period in which the actuation of the ATWS-RPT instrumentation is needed is very low. This minimizes the potential for plant transients that can occur during shutdown by providing additional time to restore the trip function prior to initiating a shutdown. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The ATWS-RPT trip functions are designed to trip the recirculation pumps following events where a scram does not (but should) occur, to lessen the effects of an ATWS event. The one hour period allows sufficient time for the operator to take corrective actions, thus minimizing the potential for plant transients that can occur during shutdown by providing additional time to restore the trip function prior to initiating a shutdown. As such, any reduction in a margin of safety by the added allowed outage time will be offset by the benefit gained by avoiding an unnecessary plant shutdown transient. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.5.1 - ECCS INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow an inoperable channel to be placed in the tripped condition to satisfy the Required Actions and allow continued operation. A tripped channel continues to provide the required safety function. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the safety functions continue to provide the required ECCS actuation capability, including single failure conditions.

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will reduce the Applicable conditions for OPERABILITY to those conditions where the automatic closure of the recirculation system discharge valves is required by the safety analysis. The purpose of this Function is to close the recirculation system discharge valves to ensure LPCI injects into the proper location, thus ensuring backflow does not occur (backflow could result in core cooling being threatened). The LPCI continues to provide all required safety functions. Once a recirculation system discharge valve is closed, the instruments effectively have performed their function. The valves can only be reopened under administrative controls, after performing a rigorous sequence of steps. Thus, inadvertent reopening of the valves is highly unlikely. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the safety functions continue to provide the required LPCI water source whenever it may be necessary.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.5.1 - ECCS INSTRUMENTATION

"L3" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change (Proposed BFN ISTS 3.3.5.1, Required Actions B.1, B.2, C.1, D.1, E.1, F.1, and G.1) provides a one hour Completion Time from discovery of loss of initiation capability to declare the supported feature inoperable. The proposed actions do not require the system or component to be declared inoperable unless initiation capability is lost. This is acceptable since the Function is still capable of performing its design function. CTS Table 3.2.B requires the supported feature to be declared inoperable immediately when more than one trip system or the first column of Table 3.2.B is reduced by more than 1. The change does not result in any hardware or operating procedure changes. ECCS instrumentation are not assumed to be initiators of any analyzed event. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L3" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Currently, no specific time period is allowed to declare a supported feature inoperable upon discovering a loss of initiation capability. The proposed change allows one hour. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping channels. The proposed actions do not require the system or component to be declared inoperable unless initiation capability is lost. This is acceptable since the Function is still capable of performing its design function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.5.2 - RCIC SYSTEM INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow an inoperable channel to be placed in the tripped condition to satisfy the Required Actions and allow continued operation. A tripped channel continues to provide the required safety function. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the safety functions continue to provide the required RCIC actuation capability, including single failure conditions.

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change (Proposed BFN ISTS 3.3.5.2, Required Action B.1 provide a one hour Completion Time from discovery of loss of initiation capability to declare the supported feature inoperable. The proposed actions do not require the system or component to be declared inoperable unless initiation capability is lost. This is acceptable since the Function is still capable of performing its design function. CTS Table 3.2.B requires the supported feature to be declared inoperable immediately when more than one trip system or the first column of Table 3.2.B is reduced by more than 1. The change does not result in any hardware or operating procedure changes. RCIC instrumentation is not assumed to be initiators of any analyzed event. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in initiation capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L2" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Currently, no specific time period is allowed to declare a supported feature inoperable upon discovering a loss of initiation capability. The proposed change allows one hour. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping channels. The proposed actions do not require the system or component to be declared inoperable unless initiation capability is lost. This is acceptable since the Function is still capable of performing its design function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.6.1 - PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change (Proposed BFN ISTS 3.3.6.1, Required Action B.1) provides a 1 hour Completion Time from discovery of loss of isolation capability prior to requiring further action. Further action is not required by B.1 unless isolation capability is lost. This is acceptable since the Function is still capable of performing its design function. The current wording of Note 1 to both CTS Table 3.2.A and B could require further action be initiated immediately even if isolation capability is maintained. The change does not result in any hardware or operating procedure changes. ECCS instrumentation are not assumed to be initiators of any analyzed event. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in isolation capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L1" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Currently, no specific time period is allowed to initiate further required action upon discovering a loss of isolation capability. The proposed change allows 1 hour. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping channels. The proposed action does not require further action be initiated within 1 hour unless isolation capability is lost. This is acceptable since the Function is still capable of performing its design function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

"L2" CHANGE

TVA has concluded that operation of BFN in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware or operating procedure changes. PCIS Functions are not assumed to be initiators of any analyzed event. The change will not allow continuous operation such that a single failure will preclude the affected isolation function from being performed. This change allows an additional 4 hours to close the MSIVs and an additional 12 hours to reach MODE 4, which provides a reasonable amount of time to perform an orderly closure of the valves (which requires entry into MODE 2) or a shutdown, thus further minimizing a potential upset from a too rapid decrease in plant power. Additionally, the consequences of an event occurring while the unit is reducing power in order to close the MSIVs during the extra 4 hours or is being shutdown during the extra 12 hours is the same as the consequences of an event occurring during the current time periods. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   The increased time allowed for closing the MSIVs or reaching MODE 4 with inoperable channels is acceptable based on the small probability of an event requiring the inoperable channels to function and the desire to minimize plant transients. The 4 hour and 12 hour extension will provide sufficient time for the unit to close the MSIVs or reach MODE 4 in an orderly manner. As a result, the potential for human error will be reduced. As such, any reduction in a margin of safety will be insignificant and offset by the benefit gained from providing sufficient time to close the MSIVs or reach MODE 4, thus avoiding potential plant transients from attempting to close the MSIVs or reach MODE 4 in the current time.

"L3" CHANGE

TVA has concluded that operation of BFN in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will remove requirements for operability of the RHR shutdown cooling system isolation on low reactor vessel water level during MODES 1 and 2. During MODES 1 and 2, the Reactor Steam Dome Pressure-High Function is required to be OPERABLE and will maintain the RHR SDC Supply isolation valves isolated when there is significant pressure in the Reactor. Additionally, during MODES 1 and 2 the Drywell Pressure-High Function will provide a PCIS isolation signal to the RHR shutdown cooling valves. The RHR LPCI to Reactor isolation valves are essential for post accident mitigation, and therefore, the PCIS signal to these valves is only enabled when in shutdown cooling. In addition, the system is administratively controlled to prevent unexpected loss of inventory via these flow paths while in these MODES. Thus, the Reactor Steam Dome Pressure-High and Drywell Pressure-High instrumentation provide acceptable single failure proof protection in MODES 1 and 2 for mitigation of any accidents previously analyzed. Therefore, this change does not significantly increase the probability of consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the requirements continue to provide acceptable capability of the assumed function's purpose (to isolate the RHR SDC System) under these conditions.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.6.1 - PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

"L4" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow continued operation with inoperable channels if the affected penetration is isolated. Isolated penetrations are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, isolating the penetration fulfills the post- accident function of the isolation logic. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the required safety function of the inoperable channels will be fulfilled.

"L5" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow continued operation with inoperable channels if the affected penetration is isolated. Isolated penetrations are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, isolating the penetration fulfills the post-accident function of the isolation logic. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the required safety function of the inoperable channels will be fulfilled.

"L6" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will provide additional time to shut down the unit if main steam line isolation is not desired. Placing the unit in MODE 4 effectively removes the need for isolation. Inoperable main steam isolation logic is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Also, this change does not further degrade the capability of the system to perform its required function under these circumstances. In addition, the consequences of an accident in the additional 24 hours provided to operate with the main steam lines open (although the unit is being shut down during this 24 hours), are the same as during the current 12 hours provided to close the main steam lines. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the additional time is minor and is consistent with the time period necessary for orderly reductions of power.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.6.1 - PRIMARY CONTAINMENT ISOLATION INSTRUMENTATION

"L7" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow an inoperable channel to be placed in the tripped condition to satisfy the Required Actions and allow continued operation. A tripped channel continues to provide the required safety function. Therefore, this change does not significantly increase the probability or consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the safety functions continue to provide the required ECCS isolation capability, including single failure conditions.

"L8" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Proposed Required Action F.1 allows 1 hour and Actions for LCO 3.6.1.3 allow four hours to isolate the affected penetration flow path(s), whereas current Technical Specification provide no specific time period. Primary Containment Isolation System instrumentation are not assumed to be initiators of any analyzed event. Therefore, the proposed change cannot result in an increase in the probability of an accident previously evaluated. The 1 hour Completion Time provides the operator sufficient time to isolate the affected penetration flow path(s) and minimizes the risk of a plant transient. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   Currently no specific time period is provided for completing required compensatory actions associated with inoperable instrumentation. The proposed change allows one hour to complete these actions. The one hour Completion Time allows the operator time to isolate the affected penetration flow path(s) without unnecessarily challenging plant systems. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.6.2 - SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   Proposed Required Actions C.1.1 and C.2.1 allow 1 hour to complete required actions, whereas current Technical Specification (Note G of Table 3.2.A) requires the reactor building be isolated and the SGT system be initiated without providing a specific time period. Primary Containment Isolation System instrumentation are not assumed to be initiators of any analyzed event. Therefore, the proposed change cannot result in an increase in the probability of an accident previously evaluated. The 1 hour Completion Time provides the operator sufficient time to establish required plant conditions or to declare the associated components inoperable without unnecessarily challenging plant systems. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   Currently no specific time period is provided for completing required compensatory actions associated with inoperable instrumentation. The proposed change allows one hour to complete these actions. The one hour Completion Time allows the operator time to establish required plant conditions or to declare the associated components inoperable without unnecessarily challenging plant systems. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.6.2 - SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION

"L2" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow continued operation with inoperable channels provided the associated zones have been isolated and the SGT System started. Isolated zones (i.e., penetrations) and operating SGT subsystems are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, isolating the penetration and starting the SGT System fulfills the post-accident function of the isolation logic. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the required safety function of the inoperable channels will be fulfilled.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.6.2 - SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION

"L3" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow continued operation (i.e., refueling) with inoperable channels provided the associated zones have been isolated and SGT System started. Isolated zones (i.e., penetrations) and operating SGT subsystems are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, placing the systems in the accident condition fulfills the post-accident function of the isolation logic. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the required safety function of the inoperable channels will be fulfilled.

"L4" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change is less stringent since it deletes the requirement to demonstrate a redundant logic is operable when a logic trip system is found inoperable. The normal test frequency for equipment in this Specification continues to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does 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 proposed change does impose different requirements. However, these changes are not related to any assumptions made in the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   Increased testing of redundant components when one component is inoperable has not been shown to detect other inoperable components any better than testing at the normal SR test interval. The use of plant controlled programs to find common cause failure modes and the new Safety Function Determination Program in BFN ISTS 5.5.11 will provide necessary assurance of system operability. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.6.2 - SECONDARY CONTAINMENT ISOLATION INSTRUMENTATION

"L5" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change (Proposed BFN ISTS 3.3.6.2, Required Action B.1) provides a 1 hour Completion Time from discovery of loss of isolation capability prior to requiring further action. Further action is not required by B.1 unless isolation capability is lost. This is acceptable since the Function is still capable of performing its design function. The current wording of Note 1 to both CTS Table 3.2.A could require further action be initiated immediately even if isolation capability is maintained. The change does not result in any hardware or operating procedure changes. ECCS instrumentation are not assumed to be initiators of any analyzed event. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities. The 1 hour Completion Time from discovery of loss in isolation capability is acceptable because it minimizes risk while allowing time for restoration or tripping channels. Therefore, the probability or consequences of an accident previously evaluated is not significantly increased.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L5" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Currently, no specific time period is allowed to initiate further required action upon discovering a loss of isolation capability. The proposed change allows 1 hour. The 1 hour Completion Time is intended to allow the operator time to evaluate and repair any discovered inoperabilities and is acceptable because it minimizes risk while allowing time for restoration or tripping channels. The proposed action does not require further action be initiated within 1 hour unless isolation capability is lost. This is acceptable since the Function is still capable of performing its design function. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

"L6" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change allows the required logic components in the Reactor Building Isolation (refueling floor), Reactor Building Isolation (reactor zone), SGTS Train A, SGTS Train B, and SGTS Train C Logic Systems to be functionally tested with other associated logic every 18 months, whereas current Technical Specification (CTS Table 4.2.A) requires a functional test of this logic every 6 months. The logic is not assumed to be an initiator of any analyzed event. Therefore, the proposed change cannot result in an increase in the probability of an accident previously evaluated. The proposed change only extends the LOGIC SYSTEM FUNCTIONAL TEST frequency for portions of the required logic and does not reduce the OPERABILITY requirements for equipment for mitigation of the consequences of an accident (i.e., does not reduce the required number of instruments, component performance requirements, etc.). Since equipment OPERABILITY requirements are not reduced and the 18 month frequency for performance of the LOGIC SYSTEM FUNCTIONAL TEST is considered appropriate, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change only affects the frequency of performance of the LOGIC SYSTEM FUNCTIONAL TESTS, and thus, does not introduce any new mode of plant operation. Therefore, this change does not create the possibility of a new or different kind of accident from any previously evaluated.

"L6" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

Currently the LOGIC SYSTEM FUNCTIONAL TESTS are required to include actuation of the actuated device where practical. In the proposed ISTS, a LOGIC SYSTEM FUNCTIONAL TEST is required to test the logic up to, but not including, the actuated device. Considering logic reliability, that the actuated devices will be tested more frequent than every 18 months in proposed ISTS Sections 3.6.4.2 and 3.6.4.3, and that a LOGIC SYSTEM FUNCTIONAL TEST frequency of 18 months allows performance of the surveillance under the conditions that apply during a plant outage which precludes the potential for an unplanned transient with the reactor at power, the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.7.1 - CONTROL ROOM EMERGENCY
VENTILATION SYSTEM INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will allow continued operation with inoperable channels in both trip systems placed in the tripped condition. Tripped channels in an isolation logic are not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Further, placing the channels in the tripped condition fulfills the post-accident function of the isolation logic. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the required safety function of the inoperable channels will be fulfilled.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.8.1 - LOSS OF POWER (LOP) INSTRUMENTATION

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change does not result in any hardware or operating procedure changes. The LOP instrumentation is not assumed to be an initiator of any analyzed event. The instrumentation's role is in mitigating and thereby limiting the consequences of a design basis accident. The instrumentation actuates to ensure the Diesel Generators (D/G) are initiated to ensure power is provided to required safety systems during a design basis accident. The proposed change to the ACTIONS will not allow continuous operation such that a single failure will preclude D/G initiation from mitigating the consequences of a design basis transient. Therefore, the proposed change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed changes do not involve any design changes, plant modifications, or changes in plant operation. The system will continue to function in the same way as before the change. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   No significant reduction in a margin of safety is involved with this change since the Required Actions have been developed to assure the D/G instrumentation remains capable of mitigating the consequences of design basis accidents. This change also provides a benefit through the potential avoidance of an unnecessary plant transient when alternate compensatory measures are available to ensure the instrumentations intended function is satisfied.

"L2" CHANGE

Deleted.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.8.2 - RPS ELECTRIC POWER MONITORING

"L1" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   This change will limit the required applicability to those conditions during which the RPS electric power monitors provide a necessary function. Although loss of power is considered in conjunction with design basis accidents, it is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Also, this change does not degrade the capability of the system to perform its design basis function when needed. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. The proposed amendment does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed amendment does not involve a significant reduction in a margin of safety.

   This change does not involve a significant reduction in a margin of safety since the monitors are provided to assure adequate power is available to the RPS when required and this change only affects conditions where such power would not be required.

"L2" CHANGE

Deleted.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
BFN ISTS 3.3.8.2 - RPS ELECTRIC POWER MONITORING

"L3" CHANGE

TVA has concluded that operation of Browns Ferry Nuclear Plant in accordance with the proposed change to technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91 (a)(1), of the three standards set forth in 10 CFR 50.92.

1. The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed change does not involve any hardware changes. The RPS electric power monitoring assemblies are not assumed to be initiators of any analyzed event. The role of the RPS electric power monitoring assemblies is ensuring that the equipment powered from the RPS buses can perform its intended function, thereby mitigating and limiting the consequences of analyzed events. The RPS electric power monitoring assemblies perform this role by acting to disconnect the RPS bus powered equipment from the power supply under conditions that could damage the equipment. The proposed change, which extends the time allowed to de-energize the affected bus from 30 minutes to 1 hour, does not allow continuous operation with the RPS electric power monitoring assemblies protective function lost. The 1 hour Completion Time for de-energizing the affected bus minimizes the risk associated with the loss while allowing time to remove the inoperable electric power monitoring assemblies from service in an orderly manner. In addition, the consequences of an event occurring during the proposed Completion Time are the same as the consequences of an event occurring during the current Completion Time. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed amendment does not 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 operation. The proposed change will not allow continuous operation with the protective function of the electric power monitoring assemblies lost. The proposed change only allows a 1 hour time period in this condition before deenergization of the affected bus is required. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

"L3" CHANGE (continued)

3. The proposed amendment does not involve a significant reduction in a margin of safety.

The proposed 1 hour Completion Time for de-energizing the affected bus when both RPS electric power monitoring assemblies of a power supply are inoperable is acceptable based on the ability of the remaining RPS buses and the small probability of an event requiring the inoperable RPS electric power monitoring assemblies to protect the associated RPS bus powered equipment. Providing a 1 hour Completion Time will minimize the risk associated with the inoperable RPS electric power monitoring assemblies while allowing time to attempt restoration or de-energize the affected bus in an orderly manner. As such, any reduction in a margin of safety by providing a 1 hour Completion Time will be offset by the benefit gained from avoiding a potential plant transient initiating from bus deenergization which may cause a half scram or group isolation. Therefore, this change does not involve a significant reduction In a margin of safety.